APPENDIX B
DETAILED PROJECT INFORMATION

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B.1 Introduction

A conceptual design for the Versatile Test Reactor (VTR) has been developed to meet user-identified needs for a fast neutron flux test facility. The VTR would provide an environment in which test specimens, such as new types of reactor fuels and materials, could be exposed to high levels of neutron flux, enabling the simulation of years of neutron exposure in a power reactor in significantly less time. After irradiation in the VTR, test specimens would be examined in post-irradiation examination facilities. Test assembly examination would be performed in facilities specifically designed to safely handle radioactive materials. VTR fuel would be fabricated at existing U.S. Department of Energy (DOE) facilities where upgrades involving removal of existing equipment and installation of new equipment would be required. DOE would put in place the facilities and processes for the treatment and disposition of spent VTR driver fuel. VTR driver fuel would not be reprocessed for the recovery of special nuclear material (plutonium or enriched uranium), but instead the entire driver assembly (including upper and lower reflectors, caps, etc.) would be melted for ultimate disposal.

This appendix provides information about the design of these facilities: the VTR, test assembly post-irradiation examination facilities, feedstock preparation facilities, driver fuel fabrication facilities, and spent fuel treatment and storage facilities. It also provides information about how the activities at these facilities would be implemented at the proposed DOE sites. The VTR would be a new facility, but other activities could be performed in new facilities or at existing facilities (with or without modification).

B.2 Versatile Test Reactor

B.2.1 Introduction

The current VTR concept is a sodium-cooled, pool-type fast reactor that provides a fast neutron spectrum environment for testing advanced nuclear fuels and materials. It generates approximately 300 megawatts thermal (MWth) and would make use of the technologies incorporated into the GE Hitachi Power Reactor Innovative Small Module (PRISM) design.[1] The VTR would meet the test reactor requirements identified in the *Mission Need Statement for the Versatile Test Reactor (VTR), A Major Acquisition Project*, as shown in **Table B–1** (DOE 2018b). In addition to these reactor parameters, the selection of the reactor type and fuel type would meet the requirement for the test facility program to provide management of the reactor fuel.

Unlike the PRISM reactor, which is designed as an electrical power plant, the VTR would be used solely as a test reactor for advancing the understanding of materials and fuels that could be used in current or future reactor designs. This results in several differences in the design and operation of the VTR from the PRISM.

The VTR, like the PRISM, would be a fast reactor. A fast reactor is a category of nuclear reactor in which the fission chain reaction is sustained by fast neutrons (carrying energies above 0.1 million electron volts (MeV) to about 10 MeV and travelling at speeds of thousands to tens of thousands of kilometers per second), as opposed to thermal neutrons used to sustain the fission chain reaction in thermal-neutron reactors. A fast reactor needs no neutron moderator, but requires fuel that is relatively rich in fissile

[1] The PRISM design is an evolutionary design based on the Experimental Breeder Reactor (EBR)-II, which operated for over 30 years. PRISM received a review by the Nuclear Regulatory Commission as contained in NUREG-1368, *Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor*, which concluded that "no obvious impediments to licensing the PRISM design had been identified."

material when compared to that required for a thermal-neutron reactor.[2] Since the VTR would be designed to test fuels and other materials in a fast flux environment, the design has been selected to maximize the number of fast neutrons present in the reactor core. The core design incorporates a reflector. The reflector would consist of assemblies of material surrounding the core that reflect neutrons that travel out of the fueled (active) region of the core back into the core, without significantly slowing them down. Also, there are no materials within the reactor specifically intended to moderate (slow down) the neutrons as there are in water-cooled nuclear power reactors; moderated neutrons are effectively lost fast neutrons.

Table B–1. Versatile Test Reactor Test Requirements

Key Performance Parameter	*Target Objective*	*VTR Conceptual Design* [a]
Provide a high-peak neutron flux (neutron energy > 0.1 million electron volts) with a prototypic fast reactor neutron energy spectrum	≥ 4×10^{15} neutrons per square centimeter/second	≥ 4×10^{15} neutrons per square centimeter/second
Provide high neutron dose rate for materials testing, quantified as displacements per atom	> 30 displacements per atom/year	51 displacements per atom/year for HT-9 and other structural materials with irradiation over three 100-day cycles (17 displacements per atom/cycle).
Provide an irradiation length that is typical of fast reactor designs	0.6 meters ≤ irradiation length ≤ 1.0 meter	0.8 meter active core height
Provide a large irradiation test volume within the core region	≥ 7 liters	Individual test volumes of greater than 7 liters, in multiple test locations
Provide experiment hardware such as casks and storage locations to support experimental mission	Provide capability for open-core, closed loops, and rabbit facility for testing sodium, lead, lead-bismuth, helium, and molten salt loops	Incorporates six positions for highly instrumented test assemblies that allow testing under different coolants, and including a rabbit facility for rapid insertion/removal of a test specimen, plus additional positions for non-instrumented assemblies

HT-9 = a stainless-steel alloy of iron, chromium, molybdenum, tungsten, nickel, and carbon; VTR = Versatile Test Reactor.

[a] The VTR test requirement parameters are as identified in the VTR Conceptual Design Report (INL 2019b). As the design evolves, these parameters are subject to change. But, none would be allowed to be changed to the extent that any Key Performance Parameter Target Objective would not be met.

A sodium-cooled reactor is a type of liquid metal reactor that uses liquid sodium as the primary coolant for the reactor. Because of the physical and thermal properties of sodium, the reactor operates slightly above atmospheric pressure and with coolant temperatures of up to 1,100 degrees Fahrenheit (°F). The primary heat removal system (HRS) operating pressure is significantly lower than that of a typical commercial light water reactor, and the operating (coolant) temperature of the fuel is higher than a typical commercial light water reactor. The reactor, primary HRS, and safety systems would be similar to those of the PRISM design. However, since the VTR is a test reactor and would not be used for electrical power generation, the secondary systems would be much simpler. The heat generated during operation would be transferred from the primary HRS to a secondary coolant system. Both coolant systems would use liquid sodium as coolant. Heat would ultimately be rejected to the atmosphere through a set of sodium-to-air heat exchangers within the secondary coolant system.

The VTR would be a pool-type reactor with both a reactor vessel and a guard vessel. This designation reflects the configuration of the primary HRS. In a pool-type reactor, the components of the primary HRS

[2] In contrast, most operating commercial nuclear power plants are thermal reactors, and the fission chain reaction is sustained by thermal neutrons. Thermal neutrons are less energetic than fast neutrons (more than a million times less energetic [about 0.025 MeV] and travelling at speeds of about 2.2 kilometers per second), having been slowed by collisions with other materials such as water. The thermal neutron spectrum refers to the range of energies associated with thermal neutrons.

are physically located within the reactor vessel. In the case of the VTR, this includes the primary electromagnetic[3] (EM) pumps and the intermediate heat exchangers. There are no penetrations in the sides of a pool-type reactor vessel or the guard vessel. The secondary cooling system pipes exit the reactor through the reactor vessel head. In contrast, a loop-type reactor has vessel penetrations for primary coolant, and the major pieces of equipment for the primary HRS are located outside of the reactor vessel. The major advantages of the pool-type reactor are a reduction in the number of penetrations in the reactor vessel and an overall reduction in size of the primary cooling systems. With the use of a guard vessel, which would maintain the sodium level within the core high enough to ensure core cooling, there is a significantly reduced likelihood of a loss of cooling accident.

The VTR, like the PRISM, would use metallic alloy fuels. The conceptual design for the first fuel core of the VTR proposes to utilize a uranium-plutonium-zirconium alloy fuel. Such an alloy fuel was tested previously in the Experimental Breeder Reactor (EBR)-II, the Fast Flux Test Facility (FFTF), and the INL Transient Reactor Test Facility. Later reactor fuel could consist of other mixtures and varying enrichments of uranium and plutonium and could use other alloying metals in place of zirconium.

The VTR is being designed for an operational lifetime of 60 years.

Unless otherwise identified, the following information is taken from the Idaho National Laboratory (INL) VTR Conceptual Design Report (INL 2019b).

B.2.2 Versatile Test Reactor General Arrangement

Regardless of the location of the VTR, the physical layout of the facility is expected to be similar (see **Figure B–1)**. The design can be developed independent of the final siting of the facility. There would be four major structures associated with the VTR: the reactor building (called the Reactor Facility), the secondary heat rejection system sodium-to-air heat exchangers (SAHXs), a plant electrical switchyard, and an Operational Support Facility. Additional structures[4] would include a Perimeter Intrusion Detection and Assessment System (PIDAS) with a double fence and guard posts/access ports. The Operational Support Facility would be located outside of the PIDAS. The VTR complex would cover approximately 25 acres (INL 2020c).

The Reactor Facility would contain most of the systems and components required for operation of the reactor. At grade level, the facility would house reactor systems equipment, experiment support area, operating floor crane (bridge crane), receiving and shipping area (truck bay), access to below-grade storage for fuel casks and experiments, and the Reactor Vessel Auxiliary Cooling System (RVACS) stacks. The reactor vessel, temporary storage locations for fresh fuel[5] and irradiated test assemblies, and most of the RVACS would be located below grade (see **Figure B–2)**. Among the other areas that would be located within the Reactor Building are the control room, electrical and battery rooms, staging and storage areas, radiological waste storage, reactor auxiliary systems areas, and secondary cooling equipment areas. The Reactor Facility would have a single operating crane, capable of transferring core assemblies, fuel casks, test assemblies, and equipment throughout the facility.

[3] EM pumps use the interaction between magnetic fields generated by magnets and electric currents to induce flow in an electrically conductive liquid such as molten sodium. EM pumps can be designed with no moving parts.

[4] This set of additional structures is not all inclusive. Other smaller structures are included in the VTR conceptual design. Additionally, as the VTR design evolves the need for additional structures may be identified. It is anticipated that any such structures would fit within the VTR complex and not materially affect construction or operation.

[5] Spent fuel would be temporarily stored within the reactor vessel. Once sufficiently cool, this fuel would be placed in transfer casks and moved to the fuel storage pad pending transfer to the spent fuel treatment facility.

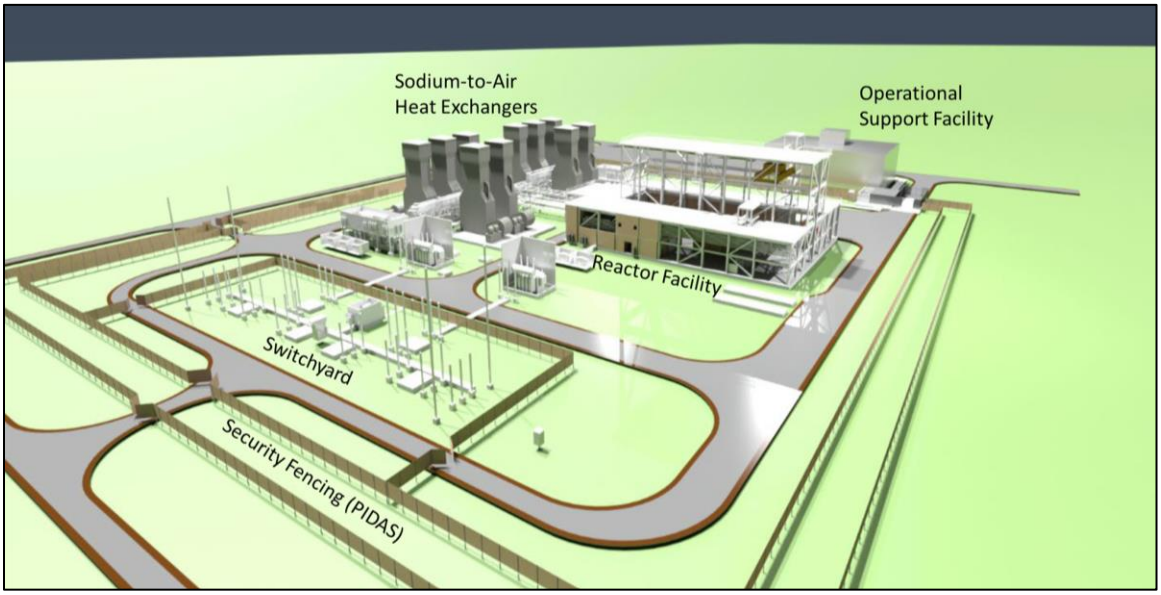


Figure B–1. Site Arrangement

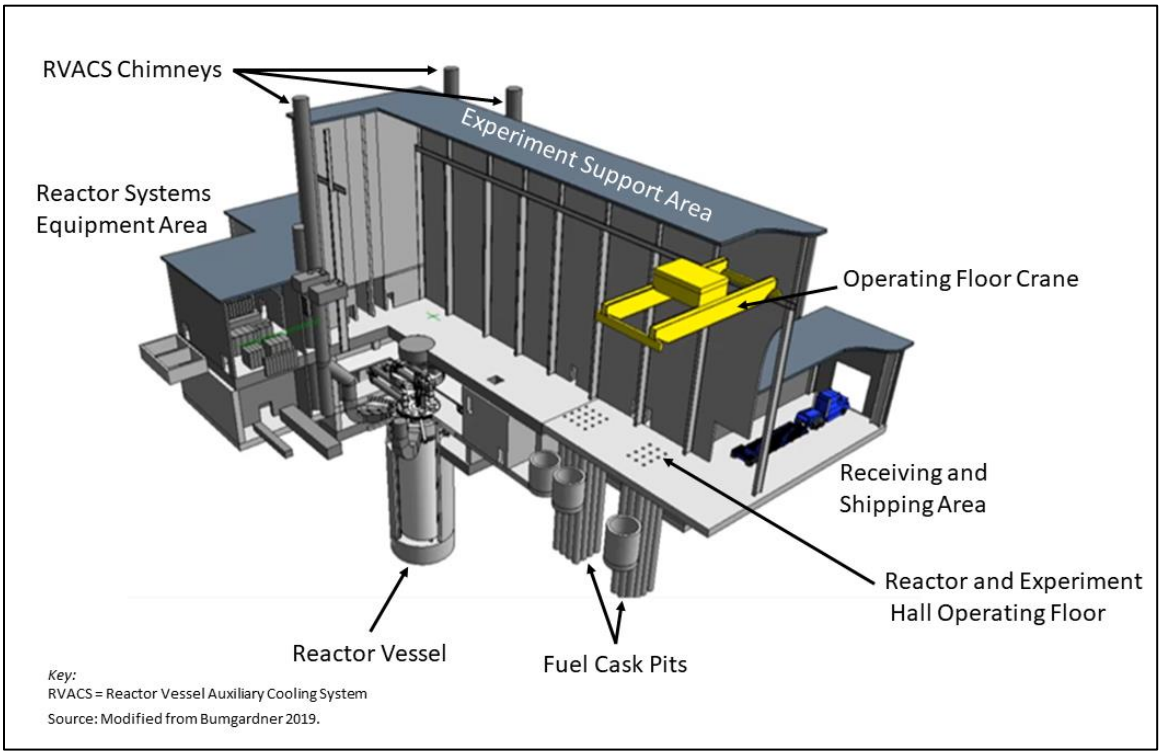


Figure B–2. Conceptual Design for the Versatile Test Reactor Facility

Most VTR activities would be performed at grade level, primarily on the Reactor and Experiment Hall operating floor. Material going into and out of the facility would pass through the shipping and receiving area. Most of the activity associated with fuel movement, spent fuel cleaning, and test assembly movement and final assembly would occur within the Reactor and Experiment Hall operating floor area. (The Reactor operating floor would be located above the reactor vessel; the Experiment Hall extends from this area to and connects with the receiving and shipping area.) The experiments support area includes locations for experiment control systems for experiments and capsule insertion and receipt areas for rabbit capsules (test capsules that can be rapidly inserted and removed from the reactor core during operation). Temporary storage areas, pits, for fresh fuel and unirradiated and irradiated test assemblies

would be located beneath the operating floor; the tops of these pits would be at the floor level of the operating floor.

Approximate physical dimensions of the Reactor Facility and a listing of the equipment located at each level of the facility are provided in **Table B–2**.

Table B–2. Versatile Test Reactor Facility Physical Dimensions

VTR Facility Level	*Dimensions Length by Width (in feet)/ Area (in square feet)*	*Equipment*
Footprint		
16 to 88.5 feet above grade	280 × 180	HVAC equipment, secondary cooling system equipment rooms, RVACS stacks, operating floor crane, gaseous radwaste equipment and stack, stairs and elevators
At grade to 16 feet above grade	280 × 180 [a]/ 42,000	Main operating floor, shipping and receiving, experiment support areas, control room, secondary cooling system equipment, electrical and battery rooms, HVAC equipment, RVACS stacks, stairs and elevators, solid radwaste storage
0 feet to 29 feet below grade	280 × 160 [a]/ 39,000	Reactor head access area, fuel cask and temporary irradiated test assembly storage areas, radiological waste storage areas, secondary cooling system equipment rooms, experiment support areas, electrical and battery rooms, building HVAC equipment, RVACS stacks and ductwork, stairs and elevators
Below grade from 29 to 41 feet	250 × 60 [b]/ 15,000	Reactor vessel, fuel cask and temporary irradiated test assembly storage areas, secondary coolant system equipment (coolant drain tanks) area, RVACS cold air plenum and ductwork, ladders
Below grade from 41 to 93 feet	31 diameter/ 750 area	The reactor vessel and enclosure (enclosure floor is at -93 feet), RVACS collector cylinder, sodium fire suppression collection tanks, sump pumps
Height (feet)		
Main building	88.5	
Annex	36	
RVACS chimneys	98	Height of the 4 chimneys (hot air exhaust elevation)
	56	Cold air intake elevation

HVAC = heating, ventilation, and air conditioning; RVACS = Reactor Vessel Auxiliary Cooling System; VTR = Versatile Test Reactor.

[a] Structure is not rectangular. Dimensions are for the longest and widest portions of the structure.

[b] The below-grade building structure would be approximately 150 × 60 feet. Fuel and test assembly storage pits comprise the remainder of the area.

The secondary HRS structures would consist of approximately 10 individual SAHXs and auxiliary equipment (e.g., SAHX fans). These SAHXs would be similar to those used for the FFTF. Heat generated by the reactor core during operations would be transferred to the HRS from the primary sodium coolant system within the reactor vessel. Pumps located within the Reactor Facility would circulate the secondary coolant (sodium) from the reactor vessel to the SAHXs. SAHX fans would dissipate heat to the atmosphere.

The Operational Support Facility would contain three floors. During construction of the VTR facility, construction workers would use the facility for office space, and a high-bay area would be used as a fabrication facility and serve as a warehousing area. Following construction completion, all three floors

would be refinished with drywall, ceilings, office cubicles, and office furniture for approximately 200 full-time staff. The building heating, ventilation, and air conditioning (HVAC) would be housed above the third floor. A reactor plant simulator would be installed to support initial commissioning and operations on the second floor of this facility. The high-bay facility area would be used to support maintenance activities and serve as a clean parts storage area. A parking lot located nearby would accommodate approximately 200 parking spaces.

B.2.3 Versatile Test Reactor Core and Fuel Design

The VTR core would consist of three regions: the fuel, reflector assemblies, and shield assemblies (see **Figure B–3**). Within the fuel region, the active part of the core, there would be driver fuel assemblies, control and safety assemblies, and test assembly locations. (Test assemblies are discussed in Section B.2.4.) The reactor core achieves peak fast neutron fluxes greater than 4×10^{15} neutrons per square centimeter per second for neutron energies greater than 0.1 MeV inside of multiple core locations for experiment items. Experiments (i.e., test specimens) would be placed in test locations in the active reactor core and in test pins located in driver fuel. Additionally, non-instrumented test locations could be located in the first row of reflector assemblies.

Core Components

Driver (fuel) assembly located in the active region of the core contains the fuel needed to power the reactor and produces the fast neutron flux necessary for irradiation of test assemblies or specimens.

Reflector assembly surrounds the active central region of the core that contains driver assemblies and test assemblies and contains material to reflect neutrons back into the central part of the core.

Shield assembly is positioned outside of the reflector assemblies within the core and contains material to absorb neutrons that pass through the reflector to reduce neutron damage to the reactor structural components.

Test assembly contains the test specimen and any equipment needed to support the experiment. Instrumented test assemblies could be as long as 65 feet and are located in the active region of the core. Non-instrumented assemblies would be the same length as driver assemblies (less than 13 feet) and may be located in either the active region of the core or in the first row of reflector assemblies.

Test specimen is the material being exposed to a fast neutron flux to determine the effects of the exposure and includes any capsule necessary to support the test. The test specimen can be no more than about 31 inches long.

Control assembly provides the core startup control, power control, burnup compensation, and absorber run-in in response to demands from the plant control system. In conjunction with safety assemblies, provide a rapid shutdown capability.

Safety assembly provides redundant rapid shutdown capability.

Core

The conceptual design for the VTR core contains 66 driver fuel assemblies within the active core. Each assembly would contain 39.9 kilograms of uranium and plutonium for a total core fuel loading of approximately 2.6 metric tons (INL 2019a). The nine safety and control assemblies would contain fuel poisons (neutron absorbers). There would be six instrumented test locations within the core. These test locations could contain instrumented fuel or material test assemblies, rabbit facility (a rapid transport system for insertion and extraction of specimens or samples during a VTR irradiation cycle), or instrumented cartridge loop assemblies. Non-instrumented experiments (i.e., test specimens) could be placed in multiple locations in the reactor core or in the reflector region. **Table B–3** summarizes these core design features.

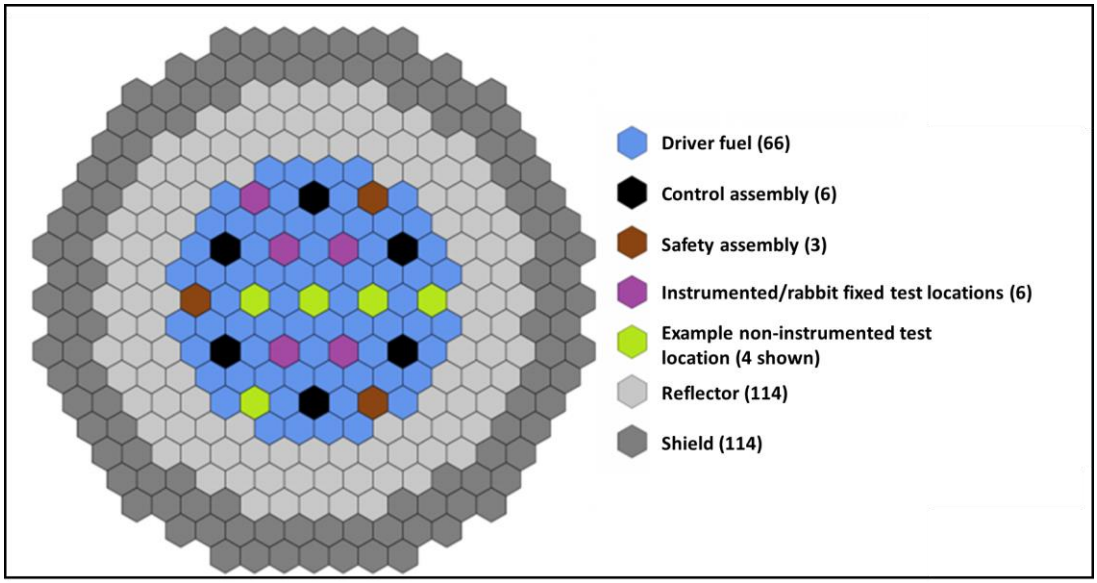


Figure B–3. Versatile Test Reactor Core Configuration

Table B–3. Key Design Characteristics of Versatile Test Reactor Core

Core Design Parameter	*Value*
General Conditions	
Pins per assembly	217
Number of driver fuel assemblies	66
Number of test assembly locations	Six fixed instrumented test locations and multiple options for non-instrumented locations in the core and reflector.
Available test volume	greater than 7 liters per test assembly location
Number of control and safety assemblies	9 (6 control and 3 safety)
Total number of fuel pins in core	14,322
Core diameter [a]	2.35 meters
Core heavy metal mass [b]	2.6 metric tons
Number of reflector assemblies [c]	114
Number of shield assemblies [d]	114
Pin Conditions	
Fuel pin length	165 centimeters
Fuel length	80 centimeters
Sodium height (above fuel)	2 centimeters
Argon height (above sodium)	80 centimeters
Pin diameter	0.625 centimeters
Fuel slug diameter	0.455 centimeters
Assembly Conditions	
Inter-assembly gap	0.3 centimeters
Duct width outside (flat to flat)	11.7 centimeters
Fuel assembly length	3.85 meters

[a] The core diameter includes fuel/test assemblies, reflector assemblies, and shield assemblies. The active core diameter (fuel and test assemblies only) would be between 132 to 144 centimeters (INL 2019a).

[b] Total uranium and plutonium mass for the initial core load.

[c] Some assembles within the inner ring of reflector assemblies could be replaced with non-instrumented test assemblies.

[d] The outer ring of shield assemblies could be replaced with spent fuel assemblies. This would provide up to 60 spent fuel storage locations.

The driver fuel would consist of hexagonal assemblies, with each assembly containing 217 HT-9 stainless-steel clad, uranium-plutonium-zirconium alloy fuel pins (see **Figure B–4**). From the bottom to the top, the driver fuel assembly is composed of the nosepiece/inlet nozzle module, the lower shield, the fuel pin bundle, the upper shield and the upper handling socket module. An assembly duct extends from the inlet to outlet modules and contains the two shields and the pin bundle. The assembly duct, support grid, and upper and lower shields would be constructed of HT-9 stainless steel. Overall, the driver fuel assembly would be about 3.85 meters long and would measure 11.7 centimeters from one flat side to the opposite flat side.

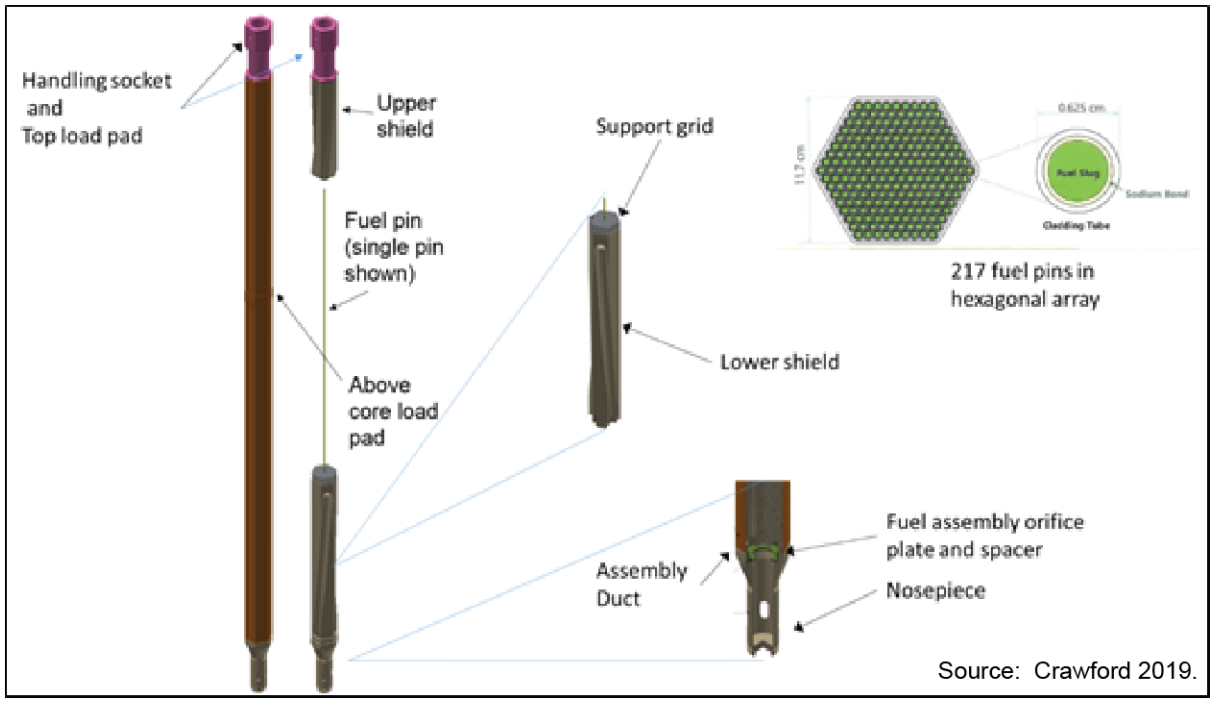


Figure B–4. Driver Fuel Assembly

The VTR core design would include six control assemblies and three safety assemblies (see **Figure B–5**). The control assemblies adjust for changes in reactivity and control the power level of the core. The safety assemblies are fully withdrawn from the fuel region during normal operation and are fully inserted into the core during reactor shutdown to provide additional shutdown margins. Each control and safety assembly is connected to a control driveline connected to a control drive mechanism, located atop the reactor upper head through penetrations in the reactor top assembly rotatable plug. All nine assemblies are configured to form a double-ducted assembly, with the inner duct containing an array of 37 wire-wrapped absorber pins. The pins are made of an HT-9 stainless-steel cladding and boron carbide (B_4C) pellets. **Table B–4** summarizes the characteristics of the control and safety assemblies.

There would be 114 radial reflector assemblies and 114 radial shield assemblies. Reflector assemblies improve neutron efficiencies (more of the neutrons generated during fission remain within the core for a longer time) by reflecting some leaked neutrons back into the core. The shield assemblies protect surrounding structures (e.g., the reactor vessel and guard vessel) from the effects of neutron radiation. Both sets of assemblies would be made with a hexagonal HT-9 stainless-steel duct.

The volume inside the reflector assembly duct would consist of HT-9 stainless-steel rods. These rods would be tightly packed (there would be no wire wrap around the rods as there would be in the driver fuel assemblies) to achieve a high steel volume. Within the reflector assembly, the HT-9 and coolant volume fractions would be 0.80 and 0.20, respectively.

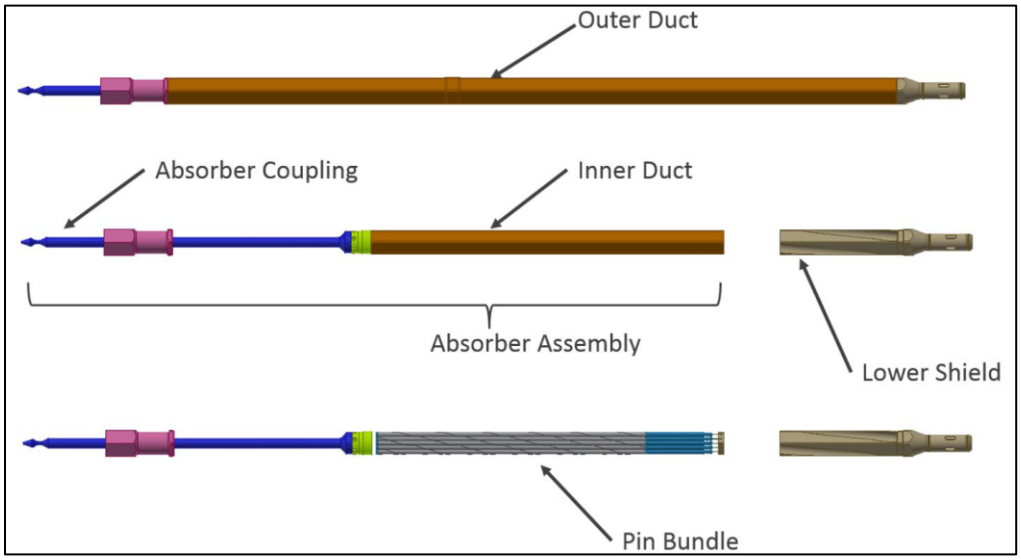


Figure B–5. Control or Safety Assembly

Table B–4. Control and Safety Rod Assembly Dimensions

Conditions	*Value*
Inter-assembly gap	3.0 millimeters
Outer hexagonal duct inside flat-to-flat distance	11.1 centimeters
Inner hexagonal duct inside flat-to-flat distance	9.9 centimeters
Number of absorber pins	37
Absorber pin outer diameter	1.54 centimeters

The shield assembly ducts would contain a bundle of wire-wrapped absorber pins made of an HT-9 stainless-steel cladding and B_4C pellets. Within the shield assembly, the B_4C absorber, HT-9, coolant, and bond gas volume fractions would be 0.40, 0.28, 0.24, and 0.08, respectively.

Driver Fuel

Both metallic and mixed oxide fuel were considered for the VTR. Metallic fuels provide several advantages over oxide fuel and were identified as the preferred fuel option. Advantages of metallic fuels over oxide fuels include:

- A smaller core at the same neutron flux level due to the higher density of fissionable metals (uranium and plutonium),
- Better performance under accident conditions,
  - Lower likelihood of energetic events that could threaten the reactor vessel and containment boundaries during core meltdown
  - Better response during a transient without scram
- Consistent performance over a wide range of fuel enrichments and alloy compositions, and
- Greater experience base with metallic fuels for fast reactors (EBR-II, Fermi-1) providing support for the licensing basis for the fuel and reactor (TerraPower 2019).

DOE considered several fuel compositions of plutonium and uranium to fuel the VTR. DOE determined that for a 300-MWth VTR, a U-20Pu-10Zr fuel with the uranium enriched to 5 percent provides the highest combination of peak neutron flux (about 4.5×10^{15} neutrons per centimeter squared per second) and technical readiness. It is the most likely fuel combination to be used in the initial fuel loading for the VTR. By weight, this fuel is 70 percent uranium enriched to 5 percent uranium-235, 20 percent plutonium, and

10 percent zirconium. The total amount of heavy metal (uranium and plutonium) required annually, as shown in **Table B–5**, for the VTR would be about 1.8 metric tons.[6] The initial fuel loading for the VTR would require about 2.6 metric tons of heavy metal (uranium and plutonium) (INL 2019a).

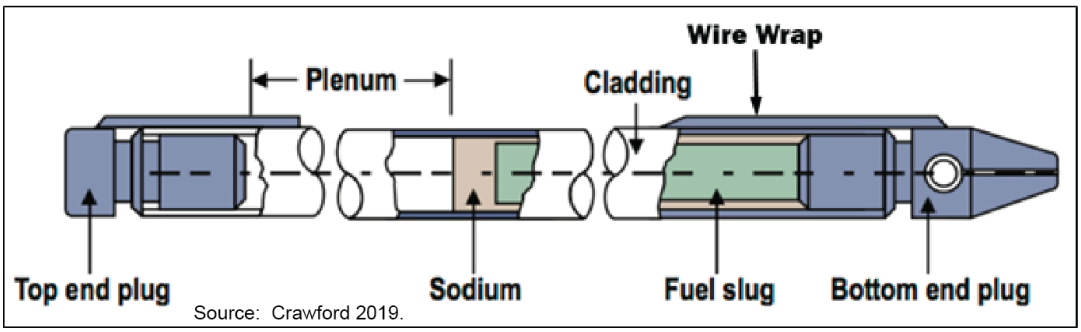
Table B–5. Versatile Test Reactor Fuel Requirements

Fuel Component	*Initial Core (kilograms)*	*Annual Requirement (kilograms)*	*Lifetime – 60 Years (metric tons)*
Plutonium	590	400	24
Uranium	2,000	1,400	85
Zirconium	290	200	12
Total Heavy Metal	2,600	1,800	110

Source: Derived from INL 2019a.

Several factors could impact the selection of future VTR fuel. For example, a desire to increase the fast neutron flux with an improvement in the readiness level (more mature fabrication and use) of higher content plutonium fuels could result in a decision to use higher plutonium content fuel. Other factors could result in the need to use lower plutonium content, but higher uranium enrichment fuels. For this environmental impact statement (EIS), it has been assumed that future fuel requirements for the VTR would be met using the U-20Pu-10Zr fuel anticipated to be used in the initial core.

Each fuel pin (see **Figure B–6**) would be 165 centimeters long and have an outer diameter of 0.625 centimeters. Only about 80 centimeters of the fuel pin would contain metallic fuel, approximately 184 grams of heavy metal (INL 2019a). Each fuel pin would contain fuel slugs, with a diameter of 0.455 centimeters. There would be an approximately equal length of a gas plenum, filled with argon in the proposed VTR design, above the fuel. This gas space provides a mechanism to limit pressure increases within the fuel pin. (When fuel is irradiated in a fast reactor, the metallic fuel swells as fission products are generated. Pores form throughout the fuel as it swells due to irradiation and pressure from the gaseous fission products. The fission product gases escape through these pores to this plenum in the fuel pin.) Between the fuel and the gas plenum, there would be a short length (2 centimeters) of sodium created during the VTR driver fuel production process (see Section B.5). The fuel, sodium, and gas plenums would be enclosed within HT-9 stainless-steel (a stainless-steel alloy of iron, chromium, molybdenum, tungsten, nickel, and carbon) cladding, about 0.05 centimeters thick. The space between the fuel and the cladding would be filled with metallic sodium to improve the heat transfer from the fuel to the reactor coolant through the stainless-steel cladding. The small amount of sodium initially above the fuel ensures that there would be sodium between the fuel and the cladding at all times. The wire wrap shown in Figure B–6 maintains spacing between fuel pins within the driver fuel assembly and is also made of HT-9 stainless steel. Top and bottom end plugs complete the structure of the fuel pin.



Source: Crawford 2019.

Figure B–6. Fuel Pin

[6] Based on the replacement of up to 45 fuel assemblies each year (INL 2020c).

B.2.4 Test Assemblies

Non-instrumented experiments (i.e., test specimens) could be placed in multiple locations in the reactor core or in the reflector regions, by replacing a fuel or reflector assembly. Instrumented experiments, which can provide real-time information while the reactor is operating, require a penetration in the reactor cover for the instrumentation stalk and can only be placed in any of six fixed locations. Any of these positions could be used for instrumented test vehicles; a rabbit test facility, and cartridge closed loops;[7] which can provide real-time information while the reactor is operating. At any one time only one of these six locations can accommodate a "rabbit" test facility, where samples can be inserted/removed while the reactor is in operation. The six instrumented test positions are served by six penetrations for the instrumentation stalk and have a direct connection through the reactor vessel head to monitors in the experiment support area with transfers on the rotatable plug, similar to the penetrations for the control assemblies (see Section B.2.5). In addition to the test assemblies, test pins could be located within the driver fuel assemblies. The number of instrumented test locations, plus the flexibility in the number and location of non-instrumented tests would strengthen the versatility of the reactor as a test facility.

Instrumented test vehicle designs have not been developed specifically for the VTR, but they would be developed based on test vehicle designs developed for the EBR-II and FFTF (**Figure B–7** provides a representative design). Based upon previous experience, instrumented test assemblies can incorporate many (e.g., greater than 50) instruments, including those to measure local temperatures, flowrates, pressures (including pressures inside fuel pin fission gas plena), and neutron fluxes. The three test assembly types currently envisioned for use in the VTR are:

- Normal Test Assembly (NTA)
  - NTAs would be the standard non-instrumented or passively instrumented open test assemblies that would be the same size, flat-to-flat, as the driver fuel assemblies.
  - The NTAs would use the same path and equipment as driver fuel for insertion and removal from the reactor.
  - These experiments would be fuels (NTA-F) or materials (NTA-M).
- Extended Length Test Assembly (ELTA)
  - All ELTAs would extend through the reactor head, and typically would have various instrumentation leads, etc., that run to the Non-Radiation and/or Radiation Experiment Rooms adjacent to the Head Access Area.
  - The ELTAs would have specialized casks capable of preheating using downward flowing argon; providing power, as necessary, to the ELTA (e.g., for cartridge loops); and the required lifting fixtures.
  - ELTAs would include fuels (ELTA-F) or materials (ELTA-M) or can be cartridge loops (ELTA CL) that could contain coolants separate from the primary sodium. Figure B–7 provides a representative design of an ELTA-M.
  - The rabbit thimble that would go into the primary coolant would be handled by the same pathway as the ELTAs, although the rabbit thimble is not considered to be an ELTA, but would use the same infrastructure for insertion and removal.
- Rabbit Test Assembly (RTA)
  - The RTA would use a capsule that contains the experiment specimens, which would be propelled down the rabbit tube into the rabbit thimble, irradiated, and recovered during or between test cycles.

[7] Non-instrumented test assemblies could also be placed within an instrumented location.

- The RTA capsules would be loaded and removed from a shielded transfer station in the Radioactive Experiment Room adjacent to the Head Access Area.
- The RTA capsule would be very specialized with tight tolerances to ensure compatibility with the rabbit thimble, fins for heat rejection if needed, and would be qualified as an experiment containment boundary. The capsule typically would contain very small samples which would nearly always be materials due to the extremely rapid insertion, which could result in a significant short reactor power disturbance for fueled specimens/tests.

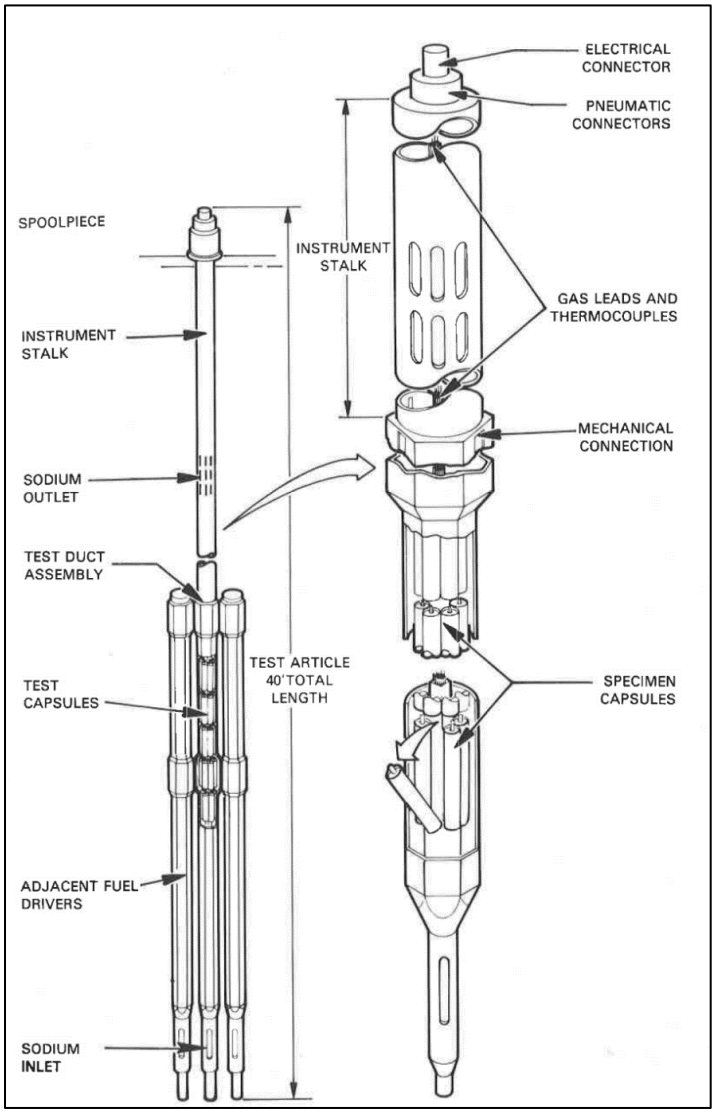


Figure B–7. Representative Instrumented Test Assembly

An important capability for the VTR would be the capability to irradiate cartridge closed loops (see **Figure B–8**) with different closed-loop coolants such as molten lead, molten salt, helium, or even sodium at different conditions than the VTR primary sodium. Thus, the VTR can directly support the development of lead- and lead-bismuth eutectic-cooled fast reactor, molten salt reactor, fluoride high-temperature reactor, high-temperature gas-cooled reactor, and advanced sodium-cooled fast reactor designs. Cartridge loop experiments have been successfully used in other test reactors; designs for VTR-specific closed loop test assemblies able to handle different coolants are under development.

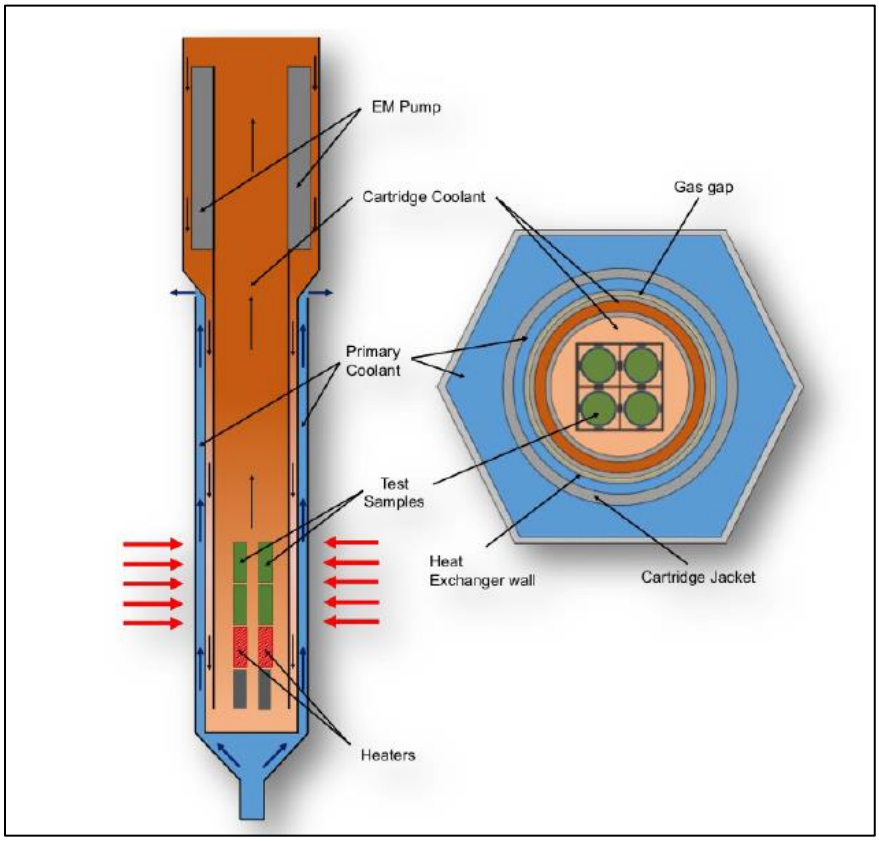


Figure B–8. Closed-Loop Cartridge Test Assembly

In the VTR, the closed-loop coolant would flow upward through a closed-loop fuel region and downward through a surrounding downcomer, where heat would be rejected through a double-wall pressure boundary to upward-flowing VTR primary sodium. Thus, the VTR primary sodium would be the heat sink for the cartridge closed loop. The cartridge closed loop may be similar in height to a driver fuel assembly and coupled to an overlying stalk with instrument leads (including leads for instrumentation to monitor the coolant purity), gas lines (some providing the ability to alter the coolant chemistry to reduce or eliminate corrosion of cladding and structures), and power cables. Each cartridge closed-loop design could incorporate an EM pump or a mechanical pump or gas circulator coupled to a motor atop the stalk through a magnetic coupling.

The remaining test locations within the core and reflector would be used for non-instrumented test assemblies.[8] Non-instrumented experiments (i.e., test specimens) could be placed in multiple locations in the reactor core or in the reflector regions, by replacing a driver fuel assembly, instrumented assembly, or reflector assembly. The non-instrumented test vehicles would be fuel assemblies used to test alternative fuel concepts (possibly a lead test assembly), cladding, and structural materials that may differ from the fuel assemblies. These test assemblies would maintain the same outer dimensions as any fuel assembly. The non-instrumented test vehicle may contain passive instrumentation (e.g., melt wires). Closed-loop cartridges would be used only in instrumented locations; all non-instrumented assemblies would be open and the VTR primary sodium would be the coolant.

[8] Generally, the number of non-instrumented test locations are 4 in the core and an additional 10 in the reflector. However, the number of non-instrumented test locations relies upon the specific cycle-dependent physics and safety calculations. In any given test cycle the number of non-instrumented test assemblies could be more or less than these estimates.

B.2.5 Reactor Vessel and Primary Heat Transport System

The VTR would be a pool-type reactor (see **Figure B–9**), so there are no primary coolant loops external to the reactor vessel. Sufficient space would be provided within the reactor vessel for the reactor core, components of the Primary Heat Transport System (PHTS) and spent fuel storage. The stainless-steel reactor vessel would be cylindrical, approximately 55.8 feet tall with a diameter of approximately 18.7 feet. The reactor vessel would be enveloped by a steel guard vessel, which envelopes the primary vessel and collects sodium in case of a leakage of the primary vessel. The guard vessel surrounds the reactor vessel and extends from beneath the reactor vessel to the upper head/top plate assembly. The space between the two vessels would be filled with argon. Attached to the top of the reactor and guard vessels would be the upper head/top plate assembly. (This assembly would connect with both the reactor vessel and the guard vessel.) The vessels would be supported by horizontal beams arranged like radial spokes and partly supported by vertical beams surrounding the guard vessel. The core is supported from the bottom on a core support structure welded to supports on the inside of the reactor vessel. The reactor vessel would be located below grade within the Reactor Building (from approximately -29 feet to -90 feet) within a concrete enclosure (see Figure B–2). Additional physical parameters are provided in **Table B–6**.

The reactor vessel contains all of the liquid sodium primary coolant. Additionally, an argon cover gas plenum would fill the top of the reactor vessel. The cover gas provides a barrier between the sodium coolant and the reactor closure assembly and serves two functions. The gas plenum provides an additional barrier to atmospheric oxygen, especially during refueling. Fission gases, air, and moisture either generated in the reactor core or present in the sodium migrate to the cover gas and would be removed by a Cover Gas Cleanup System.

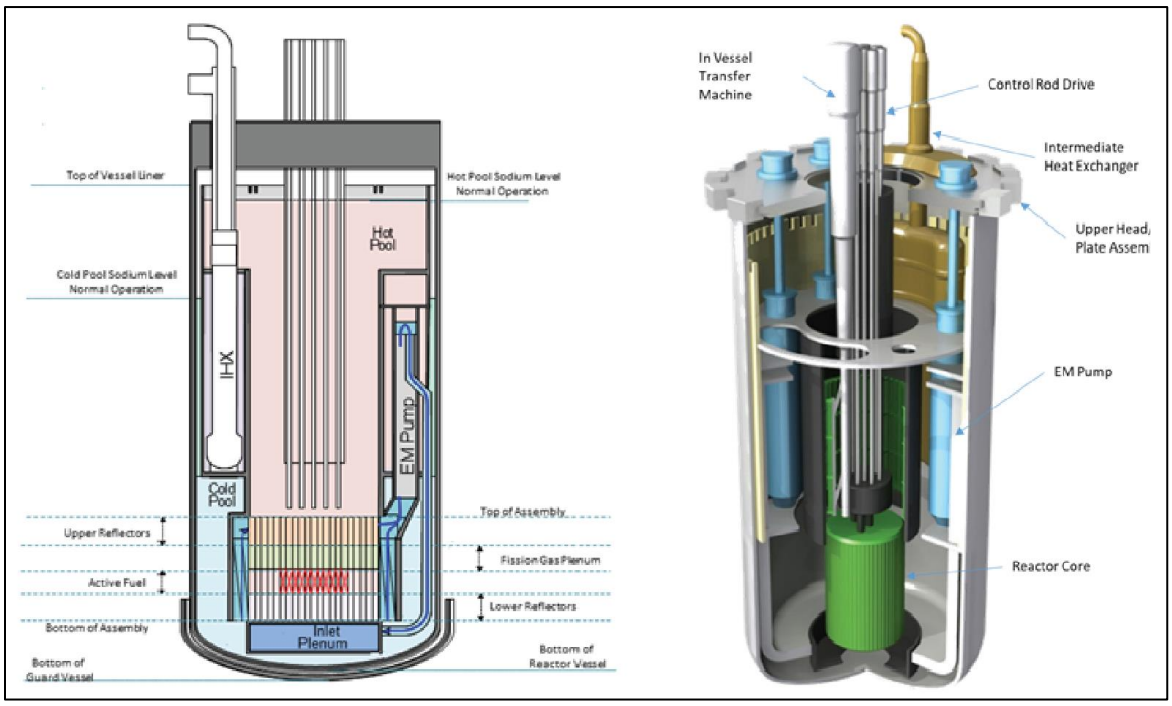


Figure B–9. Versatile Test Reactor Vessel

Table B–6. Conditions and Dimensions for the Versatile Test Reactor Primary Heat Transport System and Reactor Vessel Conceptual Design

Condition or Dimension	*Value*
Core thermal power	300 megawatts (thermal)
PHTS inlet/outlet temperatures	350/500 °C
Reactor vessel height	17.1 meters
Reactor vessel outer diameter	5.74 meters
Reactor vessel lower head outside height	1.34 meters
Guard vessel height	17.3 meters
Guard vessel outer diameter	6.04 meters
Reactor operating pressure	Slightly above atmospheric
Spent fuel storage capacity [a]	110 assemblies

°C = degrees Celsius; PHTS = Primary Heat Transport System.

[a] Spent fuel capacity includes 60 locations in the outer ring of shield assemblies and 50 locations above but outside the core diameter (at the height of the intermediate heat exchangers).

Source: INL 2019b.

There are no penetrations in the sides or bottom of the reactor vessel or the guard vessel. All penetrations are through the reactor upper head/top plate assembly which consists primarily of a reactor top plate (with a rotatable plug) and a layer of thermal insulation. Penetrations would be provided for intermediate heat exchangers (inlet and outlet flow), the primary EM pumps, the fuel handling In-Vessel Transfer Machine (IVTM), control and safety assembly drive mechanisms, experiments, core instrumentation, a maintenance access port, a transfer port, and a sodium cleanup port. The following penetrations are located in the head outside of the rotating plug; the EM pumps, intermediate heat exchangers, a transfer port, and a sodium cleanup port, **Figure B–10**.

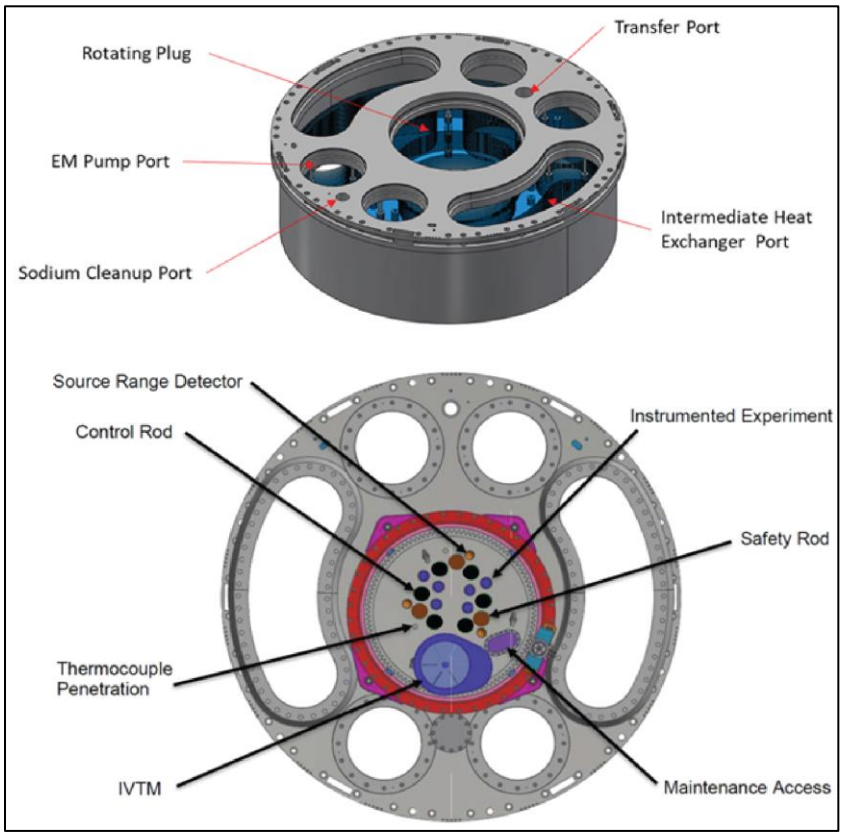


Figure B–10. Versatile Test Reactor Upper Head/Top Plate Assembly

All PHTS components would be within the reactor vessel. Major components would consist of four EM pumps and two intermediate heat exchangers, one heat exchanger for each of the two HRS secondary sodium loops. As shown in Figure B–9, the EM pumps draw sodium from the area surrounding the core within the cold pool and injects the sodium coolant through vertical piping into the inlet plenum (a space filled, in this case, with sodium) beneath the core. Coolant flows through the core to the hot pool region of the reactor vessel where it enters the intermediate heat exchangers. Heat is transferred to the secondary HRS and the primary sodium coolant returns to the cold pool portion within the reactor vessel. Primary sodium coolant pressure and temperature parameters are provided in Table B–6. The PHTS would be sized so that when the EM pumps are operating the system would be able to remove the heat generated within the reactor vessel. This includes the thermal energy of the core; energy generated by the driver fuel assemblies and test assemblies, and other heat sources including spent fuel and the thermal power deposition in the primary sodium from the EM pumps. The EM pumps and intermediate heat exchangers are mounted above the core and supported from the reactor top plate. The PHTS contains no rotating machinery such as a motor, flywheel, or generator.

The VTR reactor vessel design would allow for the storage of spent fuel within the reactor vessel. Storage of spent fuel within the reactor vessel eliminates the need for an external spent fuel storage tank. The fuel would be stored in the reactor vessel until it had cooled sufficiently to be removed from the reactor vessel and transferred to a spent fuel storage cask. Locations for the spent fuel within the reactor vessel include the outer ring of the core shielding assemblies (a spent driver fuel assembly could replace a core shielding assembly) and above and outside of the core at the level of the intermediate heat exchangers. Storage capacity for up to 110 assemblies can be obtained in this manner.

B.2.6 Heat Removal System (Secondary)

The Secondary HRS transfers heat from the PHTS to the environment. This system interfaces with the PHTS in the intermediate heat exchangers located within the reactor vessel (see Figure B–9). The system (see **Figure B–11**), would consist of two identical trains; each containing one full capacity or possibly two 50 percent capacity EM pump, a sodium expansion tank, a sodium drain tank, drain valves, a sodium purification system, and five SAHXs. The sodium drain tank, EM pumps, sodium expansion tank, and sodium purification system would have interconnecting piping located inside the rooms in the Reactor facility, and outside the building, connecting these components to the SAHXs. The design of the SAHXs would use similar concepts as those used in the FFTF secondary cooling system. Each heat exchanger would be equipped with a heater (electric or propane) to warm incoming air when needed (only at times when the VTR is shutdown) to prevent sodium freezing in the system lines (INL 2020c). System flow would be from the intermediate heat exchanger to the sodium-to-air heat exchangers to the pumps and back to the intermediate heat exchangers. Connections to the Sodium Processing System and the Cover Gas System (not shown in the figure) would be provided. System piping from within the VTR Reactor Head Access Area (but not within the reactor vessel) up to the secondary pump rooms would be double walled with the space between the walls filled with inert gas and monitored, providing an additional layer of protection between the sodium coolant and the atmosphere. HRS piping in the secondary pump rooms would have leak protections and monitoring as well. The HRS is capable of rejecting a significant amount of heat in a natural circulation mode. This passive heat rejection behavior, as well as the system providing an intact boundary, are considered safety significant functions given their role in plant defense in depth for reactor cooling in the event of a reactor trip or shutdown.

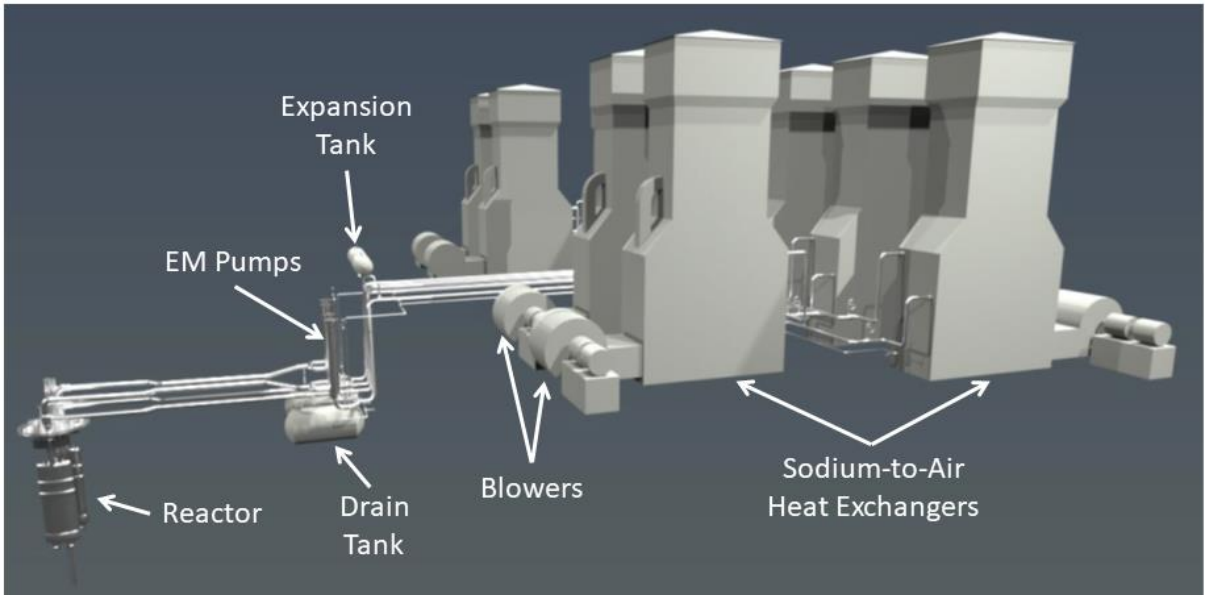


Figure B–11. Secondary Heat Removal System (One of Two Trains)

As with the PHTS, the secondary HRS would be sized to remove the required amount of heat to maintain the PHTS coolant temperature within operational limits. This includes the heat collected from the PHTS plus the thermal energy deposition from the secondary EM pumps. In addition, the PHTS and the HRS would be able to operate in conjunction in a natural circulation mode to remove reactor decay heat. Within minutes following a reactor shutdown, the heat removal capability of one of the two trains of the PHTS and HRS operating in a natural circulation mode would remove the decay heat generated by the reactor core (heat generated by the fuel, any experiments, and spent fuel stored in the reactor vessel). Therefore, sufficient heat removal capability is available to avoid significant thermal transients following a reactor shutdown. Elevation differences between the intermediate heat exchangers and the sodium-to-air heat exchangers support natural convective flow of the secondary sodium.

Table B–7 provides the coolant temperature, flow rates, and operation capacity of the system.

Table B–7. Secondary Heat Removal System Operating Parameters

Parameter	*Value*
Thermal duty (operating)	315 megawatts (thermal)
Cold leg temperature	301 °C
Hot leg temperature	462 °C
Flow per train	14,700 gallons per minute
Total flow	29,400 gallons per minute

°C = degrees Celsius.
Source: INL 2019b; GE Hitachi 2019a.

B.2.7 Reactor Vessel Auxiliary Cooling System

The RVACS would be based on the GE Hitachi PRISM RVACS design and would be a safety class, passive cooling system (no active components) that would provide decay heat removal through natural convection of air without any operator action. The RVACS would remove decay heat from the sodium pool through the reactor and guard vessel walls by radiation and convection to air outside the guard vessel. Heat would be removed to the atmosphere through the natural circulation of air due to the chimney effect. (Density differences between the cold air in the inlet and the hot air in the outlet drives the hot air up and out into the atmosphere.) The system would operate continuously, even during reactor operation. It therefore would operate in conjunction with the PHTS and HRS to remove heat during

operation. In the RVACS, air is drawn in through four chimneys, circulated around the reactor guard vessel and exits through the chimney (see **Figure B–12**). All four chimneys contain both cold-air inlet chimneys and hot-air outlet chimneys. The air outlets are located at a higher elevation than the air inlets. The RVACS would be able to perform its safety function with at least one of the four stacks out of service.

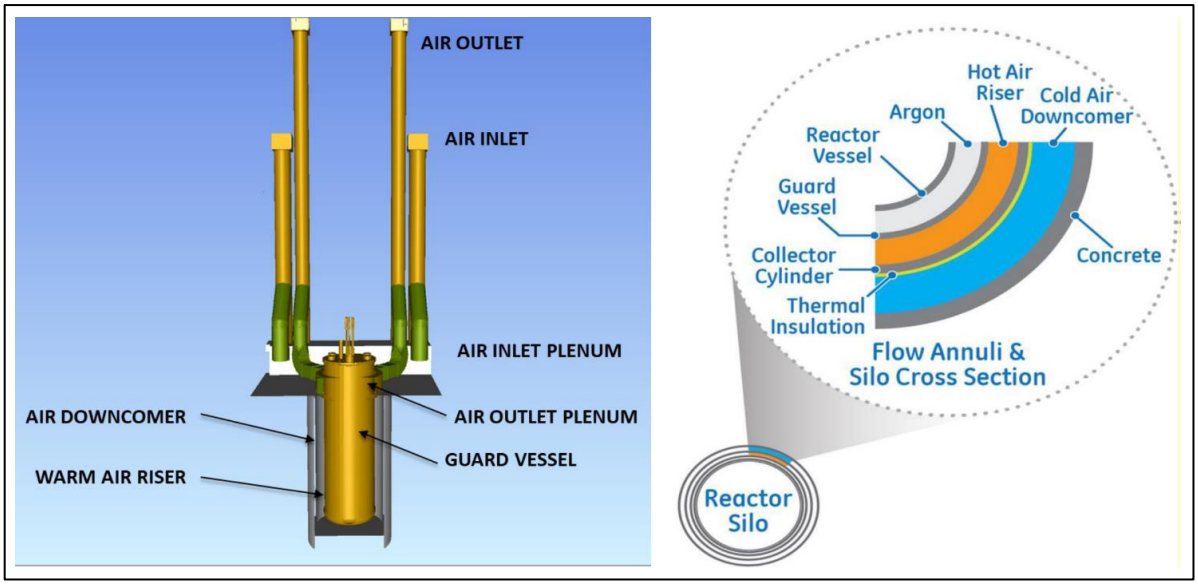


Figure B–12. Reactor Vessel Auxiliary Cooling System

The RVACS operates at a higher heat removal rate as the temperature of the primary sodium increases; alternately, as the temperature of the guard vessel outer surface decreases, so does the heat removed by the RVACS. The RVACS operates at its design capability only when it is the sole means of core heat removal, that is, only when the PHTS and secondary HRS are not functioning. The system reaches its design operation capability only after the reactor has been shut down for some period of time (on the order of a day). This means the core temperature would rise during that time before the heat removed by the RVACS would match the heat generated by the core. At equilibrium, the RVACS would remove approximately 2.8 MWth. (During power operation, the system capability would be limited to approximately 0.7 MWth).

B.2.8 Additional Systems

This section provides brief descriptions of some of the remaining VTR systems. This is not an all-inclusive set of systems (e.g., electrical systems, radiation monitoring, and control room systems are not discussed). The systems described are unique (or configured differently than in other applications) to a sodium-cooled reactor or test reactor. Additionally, the radioactive waste systems are discussed because failures associated with these systems were identified in the accident analysis as a pathway to an accidental radiological release.

Argon Gas Distribution System – The Argon Gas Distribution System would vaporize liquid argon to a suitably high pressure, filter it for removal of solid impurities, and store it under pressure as gas in a storage tank(s). An extensive distribution system of pipes, pressure regulators, and valves would deliver the argon gas to the various VTR systems and components where it would be utilized. The Argon Gas Distribution System would provide argon gas of suitable purity to:

- The reactor vessel cover gas region;
- The gap between the reactor and guard vessels;
- The cover gas regions of the secondary sodium expansion tanks and drain tanks;

- The gas space between the main and guard pipes of double-walled sodium piping;
- Driver fuel, assembly, test vehicle, and component transfer and dry storage casks; and
- Other processes requiring argon gas.

Wherever there would be a sodium system, there would be components and piping of the Argon Gas Distribution System. The system would include sodium vapor traps where needed.

Containment – The reactor and PHTS would not be enclosed inside of a containment dome structure.[9] The containment function is provided by the reactor head and the Head Access Area (see **Figure B–13**), which would be entirely below grade. Components of the Head Access Area that are part of the containment would include the area ceiling, walls, and floor; ventilation duct dampers; penetration isolation; isolation valves; and airlocks. Additionally, the outer piping of HRS double-walled piping provides containment in the event of a leak in the secondary sodium piping.

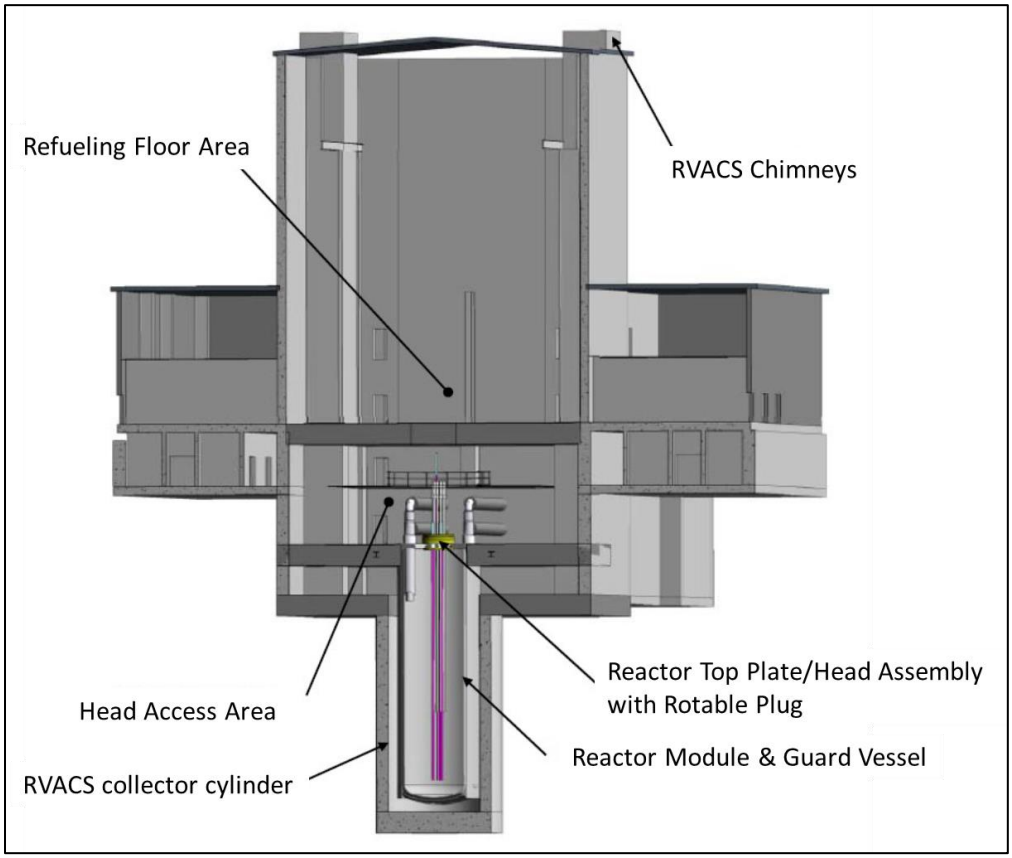


Figure B–13. View of the Versatile Test Reactor Operating Floor, Head Access Area, and Reactor

HVAC – The Reactor Facility HVAC System would provide heating, ventilation, and air conditioning for the various areas of the Reactor Facility during normal and off-normal conditions. The Reactor Facility HVAC System would also maintain humidity, pressure, and air cleanliness required for the areas served. The HVAC System would provide HVAC within the Reactor Facility by recirculating conditioned air or by once-through circulation of air. The Reactor Facility operating area and Experiment Hall and Head Access Area, as well as Reactor Facility electrical rooms would be heated and air conditioned, while other areas would

[9] The VTR operates at near atmospheric pressure. Even under post-accident conditions, reactor and containment pressures are near atmospheric. A large reinforced containment structure is not needed to prevent the release of radioactive elements to the environment under accident conditions.

be ventilated to remove heat loads with once-through circulation of air and heated with heaters, as required. Because of the potential for contamination, air from potentially contaminated spaces would be exhausted to the outside through charcoal adsorbers and high-efficiency particulate air (HEPA) filters to control the release of airborne radioactive gases and particles to the outside environment.

In-Vessel and Test Assembly Handling Systems – Movement of fuel and non-instrumented test assemblies within the reactor core would be accomplished using the IVTM. The IVTM would be used for all fuel, control, safety, reflector, and shield assemblies and non-instrumented test assembly transfer movements (except for control and safety rod movement into and out of the core) in the core, including:

- Retrieval of fresh assemblies from the transfer basket,
- Placement of fresh assemblies into the core,
- Removal of assemblies[10] from the core,
- Placement of spent driver fuel assemblies into a storage rack above the core,
- Placement of spent driver fuel assemblies into the outer row of the radial shield,
- Removal of spent driver fuel assemblies from the storage rack, and
- Placement of core assemblies in the transfer basket.

The IVTM would consist of three major parts: an upper ex-vessel drive section, a lower in-vessel section with a pantograph (a jointed framework), and a mechanical grappler. The IVTM is attached to the rotatable plug within the reactor top plate. The IVTM grappler could be positioned over any core position, over any in-vessel storage location outside of and above the core, and over the fuel transfer basket/station.

The In-Vessel Test Assembly Handling System would receive ELTA's and rabbit thimbles for transfer into and out of the reactor. This would be accomplished via a test assembly transfer cask, the building overhead bridge crane, the test assembly transfer adapter (designed to fit the test assembly ports on the rotatable plug), and the appropriate grapples and attachment mechanisms. The ELTA's/rabbit thimbles will occupy the six fixed positions provided on the rotatable plug. The In-Vessel Test Assembly Handling System would be required to:

- Raise and lock the ELTA's/rabbit thimbles into position above the core to avoid interference between test vehicles, the IVTM, and the core during refueling and experiment vehicle management;
- Unlock and lower the ELTA's/rabbit thimbles once refueling and other necessary movements are complete; and
- The ELTA's and rabbit thimbles will be designed to allow for tooling that will sever instrument cables/tubing/etc., from the ELTA's and rabbit thimbles, and the stalks can be removed, if required.

Ex-Vessel Fuel and Test Assembly Handling Systems – All fuel handling activities outside of the reactor vessel in the Reactor Facility are carried out on the Reactor and Experiment Hall operating floor, located above the reactor at grade level (see Figure B–2). The Ex-Vessel Fuel Handling System would receive fresh fuel as well as control, reflector, and shield assemblies and process spent fuel in preparation for shipment

[10] Test assemblies may be moved from the core to a storage location in the vessel to allow for decay-heat decrease before removal from the vessel. In-vessel storage would be required should the test assembly decay heat need to fall sufficiently to allow removal from the reactor vessel. All connections (power and instrumentation) would be severed before the assemblies could be moved to the storage locations. Removal of these assemblies would then be performed using the same procedure as that for any other assembly.

to a fuel treatment facility. Equipment required for Ex-Vessel Fuel Handling System operations would include:

- Assembly preheating station,
- Overhead bridge crane and assembly transfer cask,
- Fuel transfer adaptor, and
- Spent fuel washing station.

Fresh fuel would be received at the receiving and shipping area and transferred to a fresh fuel storage pit using the overhead crane. (The top of the pit is located at the floor level of the Reactor and Experiment Hall operating floor.) Prior to insertion in the core, each assembly would be transferred from the pit and placed inside a vertical preheating station filled with inert argon gas. The top of the preheating station would be at floor level of the Reactor and Experiment Hall operating floor and located near the fuel pits.

The building overhead bridge crane serving the operating floor and an overlying fuel transfer cask filled with argon would be used to transfer the fuel assembly from the preheating station to the reactor vessel. The fuel transfer cask may be capable of holding from one to three assemblies (the fuel transfer cask design has not been finalized). The bottom of the fuel transfer cask would incorporate a gate valve. A fuel transfer adaptor would be required to connect the fuel transfer cask with the fuel transport port in the reactor top plate (see Figure B–10). The fuel transfer cask would be relocated from the preheating station to the operating floor above the reactor upper head, using the building crane. Each assembly is transferred from the fuel transfer cask to the reactor using internal drives with the aid of a fuel transfer adapter. The adapter would be necessary because the fuel transfer cask is located at the refueling floor at the 0-foot elevation, and the upper head is located at the 29-foot elevation; these are connected by the transfer adapter, which would be filled with argon gas. The use of the adaptor allows for simplified movement through the Head Access Area, while protecting the reactor head and associated penetrations from potential impacts from facility cask movements. A floor valve, located at the top of the adapter, when open, would provide a conduit for lowering the assembly through the upper head and fuel transfer port into the reactor vessel transfer basket below the sodium surface.

Spent driver fuel assemblies would be removed from the reactor vessel transfer basket using the same equipment and would be transferred to a washing station located on the Reactor and Experiment Hall operating floor. Fuel pits, below the operating floor (top of pits at floor level), would be available to temporarily hold spent driver fuel assemblies after washing. The residual sodium would be removed by reacting it under tightly controlled environmental conditions and reaction rates in the washing station. The washing station top is located at floor level. A combination of nitrogen and demineralized water moisture would be used to remove sodium from the driver fuel assembly. The reaction of sodium with moisture creates hydrogen gas, as well as sodium hydroxide. The sodium hydroxide would be washed off the assembly surfaces with demineralized water. Water containing sodium hydroxide and radionuclides would be collected by the Liquid Radioactive Waste System. The assembly would be dried with heated inert gas. After washing and drying, the spent driver fuel assemblies would be loaded into transfer casks for interim storage at the fuel storage pad and eventual transfer to a fuel treatment facility. Gas containing hydrogen and radionuclides would be collected by the Gaseous Radioactive Waste System.

The Ex-Vessel Test Assembly Handling System would be similar to the Ex-Vessel Fuel Assembly Handling System. The requirements for the system would vary depending upon the content and configuration of the test assembly. However, some test assemblies would require preheating, so preheating capability would be available in some of the assembly preparation stations. The overhead crane, test assembly transfer casks, test assembly transfer adaptors (designed to fit the test assembly ports on the rotatable plug), and a test assembly washing station would all be required. Due to the length of some test

assemblies, longer than fuel assemblies (e.g., ELTA), the preparation and cleaning stations and the transfer casks for this system would be taller than those for the Ex-Vessel Fuel Handling System.

As with ex-vessel fuel movements, all test assembly handling would take place on the Reactor and Experiment Hall operating floor.

Radioactive Waste Systems (Gaseous, Liquid, and Solid) – This system has not been fully designed. The following provides a conceptual design for the Gaseous Radioactive Waste System (see **Figure B–14**). The system receives radioactive argon cover gas from the reactor, radioactive argon cover gas from sodium components, and radioactive nitrogen from washing of residual sodium off of sodium components, as well as off-gas from processes. The radioactive gas would be filtered, and radionuclides such as xenon would be adsorbed and held in charcoal filters to decay. After sufficient treatment and holding time, gaseous effluents would be passed through multiple stages of HEPA filters before being released to the environment via an exhaust stack.

The system would receive air and cover gases from all VTR building systems, including radioactive reactor cover gas, sodium component cover gas, and process off-gas. The system would be sized to support the sodium removal and decontamination of a driver fuel assembly with failed fuel or a failed experiment vehicle in addition to maintenance activities.

Radioactive gases would be initially collected in a Holdup/Sampling Tank where unfiltered gas samples could be collected. Downstream of the Holdup/Sampling Tank, the Transfer Tank, a high-pressure tank, is used to maintain a constant system pressure. Located downstream of the Transfer Tank, the treatment system would consist of two 100-percent-capacity trains containing moisture separators, upstream and downstream HEPA filters, and charcoal-adsorption delay beds. A Secondary Hold-up/Sampling Tank would be located between the filtration components and the HVAC stack and would be the point where filtered gas samples could be collected. Compressors (two 100-percent capacity between the Holdup/Sampling Tank and the Transfer Tank and two 100-percent capacity downstream of the second set of HEPA filters) would provide the motive force for gases through the system.

The Liquid Radioactive Waste System would provide for collection and processing of radioactive liquid wastes from sodium removal, decontamination, equipment and area washing, and showers/washes. Through a series of pipes and drains, the radioactive liquid wastes would be collected in collection tanks, pumped through cartridge filters (two 100-percent-capacity trains), as required by treatment facility acceptance criteria, and held up in storage tanks for export via truck to be processed outside of the VTR. The Liquid Radioactive Waste System would incorporate a demineralized water supply system. Demineralized water would be provided to the moist gas generator for removal of sodium via interaction with moist gas inside the sodium washing station and other facility users. After use, the contaminated water would be collected as part of the liquid radioactive waste. The cartridge filters would be processed as solid radioactive waste.

The Solid Radioactive Waste System would receive solid radioactive waste from the other plant systems, perform any size reduction required, package the waste, and temporarily store the waste before final export from the VTR facility. The storage area would provide one outage (25 days or less) of storage space. The system would be monitored locally to ensure operating conditions are within specified parameters and that the system is configured appropriately.

Sodium Fire Protection System – The Sodium Fire Protection System would include instrumentation/ detectors to detect sodium leaks and sodium fires, portable fire extinguishers for fighting sodium fires of limited size by personnel, and design features to mitigate against the effects of postulated bounding and conservative sodium fire scenarios.

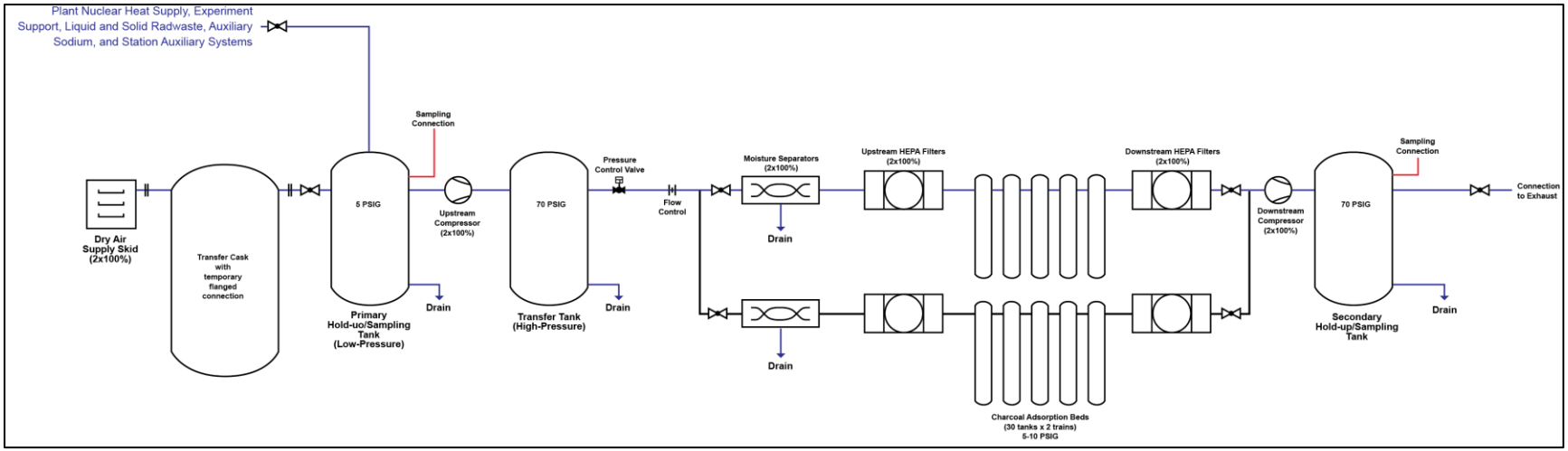


Figure B–14. Gaseous Waste Management System

Due to the low system operating pressures, any sodium leaks are expected to start as small weeping leaks in a "leak before break" failure mode. Sodium leak detectors would be provided to detect sodium leaks while they are still small, such that the affected pipe or component could be removed from service and repaired before the hole grew to a significantly larger size. For these small leaks or leaks of sodium limited in size, portable fire extinguishers containing dry powder would be provided inside areas containing sodium piping and components. If the sodium is accessible (e.g., has leaked from thermal insulation), can be observed to be burning, extinguishment is judged to be the correct action, and fighting the fire can be done safely, personnel can use the fire extinguishers to extinguish such limited fires.

In addition to the potential fire-related damage, sodium fires can result in the generation of harmful aerosols (sodium peroxide and sodium oxide) and sodium hydroxide (from chemical reaction with water) and sodium carbide (from chemical reaction with carbon dioxide), both of which are corrosive. Extinguishing a sodium fire terminates and limits the generation of these hazards.

The installation of steel catch pans or steel basins on the floor would be a mitigation design feature that would prevent released sodium from directly interacting with the concrete floor. Upon being heated, concrete could release water that would chemically interact with metallic sodium, forming hydrogen. Typically, a steel catch pan would be sized to hold more than the maximum volume of sodium that can potentially leak into a room.

As noted above, all sodium leaks are expected to start as small weeping leaks and to be detected in time such that the amount of sodium leaked remains small. However, the Sodium Fire Protection System would be designed to accommodate postulated bounding and conservative sodium-release scenarios, in which the total inventory of sodium that can potentially leak is assumed to be released.

Specific design features for preventing and mitigating sodium leaks and fires would include double-walled piping on the secondary sodium inlet and outlet main pipes from inside of the reactor Head Access Area room to the secondary pump rooms. Sodium released from a postulated leak in the main pipe would be collected in the leak-monitored and inert-gas space between the two pipes and drained into an inerted sodium collection tank, which is located inside of a vault beneath the loop sodium drain tank room. Sodium leaking outside the piping system would flow onto a catch pan on the floor with pan drains leading to the sodium collection tank. The sodium collection tank would incorporate a perforated plate with a significantly reduced area for air flow near the top, to reduce the transport of oxygen to the sodium pool surface and thereby reduce the sodium burning rate. The sodium collection tank would incorporate a vent for heated gas, would be trace heated to prevent the condensation of water moisture from air, and would enable collected sodium to be heated and melted. The piping delivering sodium to the sodium collection tank would also be trace heated to prevent sodium from freezing inside of the piping.

Sodium Purification Systems – An in-vessel Primary Sodium Purification System for the VTR is in the conceptual design phase. The Primary Sodium Purification System would remove impurities (mainly oxygen) above an established level from the PHTS sodium to maintain a desired level of purity. It also would remove radionuclides, primarily cesium, that may be released from failed fuel. The system would be a module that is installed inside the reactor vessel and would consist of two integrated purification units with a cold trap cartridge and a cesium trap cartridge. The integrated purification unit largely consists of a sodium pump, regenerative heat exchanger, non-regenerative heat exchanger, removable cartridges (to be replaced as necessary to ensure filtration capability), sodium piping, and nitrogen piping associated with the non-regenerative heat exchanger. Except for the portion which accepts insertion of a cold trap or cesium trap cartridge, the components within an integrated purification unit are largely contained within an argon-inerted and sealed vessel. To remove the necessary heat from the sodium for purification, each integrated purification unit would be associated with a closed nitrogen loop with a blower, which cools the heat exchanger and a nitrogen-to-air heat exchanger and air blower to cool the

nitrogen loop. Additional concepts for outside reactor vessel cleanup, either temporarily or permanently installed, may be explored as the design progresses.

The Secondary Sodium Purification System would remove impurities above an established level (mainly oxygen) from the secondary HRS sodium to maintain a desired level of purity. A separate purification system would be provided for each of the two secondary sodium loops. The system would also support initial fill and sodium-inventory-control operations for both the PHTS and HRS. The purification system for each secondary HRS sodium loop would be equipped with the following components:

- an EM sodium pump separate from the main-loop EM sodium pumps,
- an economizer (i.e., a regenerative heat exchanger) that partially cools sodium upstream of the cold trap via heat exchange to cooler sodium exiting the cold trap,
- a cold trap in which excess oxygen is crystallized to form sodium oxide that deposits upon a structure (e.g., a stainless-steel mesh packing) inside of the cold trap,
- a cold trap air-cooling circuit incorporating an air blower,
- a plugging temperature indicator with an air-cooling circuit,
- interconnecting piping and valves,
- instrumentation, and
- valve control actuators.

The system would receive unprocessed sodium from the secondary HRS sodium loop upstream of the main-loop EM sodium pumps and from the loop drain tank. The sodium would flow through the economizer and cold trap. The system also would incorporate piping to direct a portion of the sodium flow through the plugging temperature indicator/plugging meter. Following removal or measurement of impurities, the sodium would be returned to the secondary HRS loop at the loop expansion tank. Grab samples can be taken for analysis of the radionuclides and chemical impurities present in the sodium.

B.2.9 Operations

The nominal test-cycle length for the VTR would be 100 effective full-power days, followed by a nominal 20-day refueling outage. Driver fuel assemblies would remain in the core for a number of cycles. Those further out from the core centerline would be subjected to a lower neutron flux and undergo a slower rate of burnup. Consequently, they could be left in the core for a greater number of cycles. The goal is to achieve approximately the same mean discharge burnup in all driver fuel assemblies. A VTR driver fuel assembly may be left in the core for three, four, five, or six cycles.

The VTR test cycle would require 14 to 15 fresh driver fuel assemblies for each 100-day cycle (INL 2020c). Fresh driver fuel assemblies would be delivered by truck into the truck bay at grade level. Fresh driver fuel assemblies could be stored in fuel cask pits beneath the Reactor Operating Room floor or loaded directly into the reactor vessel. The operating area above the reactor would be a long Experiment Hall interconnected to a truck bay. The operating floor inside of the Reactor Facility would be at grade level, as shown in Figure B–2. Prior to insertion into the reactor vessel, each fresh driver fuel assembly would be properly preheated to melt the sodium to form the sodium bond with the fuel before being transferred into the reactor sodium pool. Preheating prevents thermal shock to the cold assembly when it is lowered into the sodium pool and ensures that the bond sodium in the fuel pins heats from the free surface down. Following preheating and cleaning, the assembly would be raised into a heated fuel transfer cask and moved to the reactor using the overhead bridge crane. The preheated and cleaned fresh assembly would be lowered through the fuel transport port into the transfer basket, from which it is removed and placed in the core by the IVTM.

Spent driver fuel assemblies would be transferred from the core to the spent fuel storage locations within the reactor vessel (either within the outer ring of shield assemblies or above and outside the core at the

level of the intermediate heat exchanger) using the IVTM. A spent fuel assembly would be stored in-vessel for a year or more, while its decay heat power level falls below a specified value. When sufficiently cooled, a spent driver fuel assembly would be raised from the transfer basket below the sodium level, through the fuel transport port in the reactor top plate, and placed inside a fuel transfer cask with an inert atmosphere and cooled by natural circulation. Movement of the spent driver fuel while on the Reactor and Experiment Hall operating floor has been discussed in Section B.2.8.

The overhead bridge crane would be used to move the fuel transfer cask to the sodium wash station. Residual sodium would be removed from the assembly inside of the wash station vessel by first exposing the assembly to inert nitrogen gas containing demineralized water moisture and then with demineralized water. Waste water containing sodium hydroxide and radionuclides would be collected by the Liquid Radioactive Waste System, while nitrogen containing hydrogen and radionuclides would be collected by the Gaseous Radioactive Waste System. The assembly would be dried with heated nitrogen gas and then raised up inside of an inerted dry storage/transfer cask which may hold up to six assemblies (cask design is not final). Clean and dried spent driver fuel assemblies would be transferred to a fuel storage pad for interim storage. At the storage pad, spent driver fuel assemblies would be stored in each spent fuel cask until decayed sufficiently to allow for fuel treatment, for a period of at least 3 years. Driver fuel assemblies would be stored for less than 5 years. At that time, the spent driver fuel assemblies would be transferred to a spent fuel treatment facility in preparation for ultimate storage. Spent fuel treatment and storage is discussed in Section B.4.

During refueling outages, it may be necessary to raise ELTA's and RTA's out of the core to an elevation sufficiently high above the core and lock them in the raised position to avoid interference with refueling operations. This is described in Section B.2.8, above.

ELTA, RTA, and NTA insertion and removal from the core follows a procedure very similar to that used for fresh and spent driver fuel assemblies. However, differences include:

- ELTA/RTA/NTA preparation would be required before preheating and cleaning;
- ELTA's and RTA's (up to 65 feet tall with the instrumentation stalk) require a tall test vehicle transfer cask; and
- ELTA's and RTA's would be inserted directly into the core through the test assembly penetrations in the rotatable plug, not through the transport port into a transfer basket.

ELTA's and RTA's could be:

- Removed directly from the core and transferred to a tall sodium wash station, or
- Disconnected from the assembly stalks and then moved using the IVTM to a transfer basket; and
- May be examined in Experiment Hall facilities.

The Reactor Facility layout facilitates ex-vessel test vehicle handling. The operating floor area at the reactor would be at grade level and open to a long Experiment Hall along the length of the building. The Experiment Hall would include an experiment support/preparation area. Test vehicles, particularly the ELTA's and RTA's, would be prepared in the horizontal position attached to a strongback. When ready, the test vehicle would be raised to a vertical position and placed inside of a deep pit. From the pit, the test assembly would be transferred to the reactor core in a tall transfer cask in a manner similar to that used for fresh driver fuel assemblies.

At the end of their irradiation, test assemblies would be removed from the reactor vessel and transferred to a washing station. The stalks from ELTA's and RTA's are particularly long. A separate sodium washing station, or purpose-built wash coffin connecting the fuel wash station incorporating a great height, would need to be included to remove residual sodium from stalks or complete test vehicles in which the ELTA or RTA is connected to its stalk. Movement of stalks or complete test vehicles from the rotatable plug to the

washing station would be carried out using a tall test vehicle transfer cask. Alternatively, the stalks could be removed and sectioned/cut, and washed in the fuel wash station.

At the end of their irradiation, instrumented test assemblies may have a significant decay heat similar to fuel and may require in-vessel storage while their decay heat falls. While NTA's can be handled similar to spent fuel assemblies, the stalk of ELTA's must be disconnected or severed. Once the stalk is disconnected, the ELTA would be handled in the same manner as described above for a spent fuel assembly, when being transferred to a shielded cell.

The Experiment Hall would incorporate a shielded cell located in a pit for prompt robotic post-test examination of test assemblies. The sequencing of removing residual sodium may be specific to the particular experiment and the intent of the experimenters. The ELTA stalks may be removed from the lower test vehicle portion inside of the shielded cell to make it suitable for shipment to a DOE facility for post-irradiation examination.

B.2.10 Versatile Test Reactor at the Idaho National Laboratory Site

At the INL Site, the VTR would be built adjacent to and east of the Fuel Manufacturing Facility (FMF) and Zero Power Physics Reactor (ZPPR) protected area at the Materials and Fuels Complex (MFC). The protected area PIDAS would be extended to encompass most of the VTR structures. Construction of the VTR has been estimated to take approximately 51 months, once design activities are complete. Based on the layout of the VTR (see Section B.2.2), the VTR complex at INL would occupy about 25 acres. During construction, an additional 75 acres would be required for temporary parking and equipment laydown, assembly, and staging. About 100 acres would be impacted by VTR construction (see **Figure B–15**). There is a pygmy rabbit burrow located on the southern edge of the construction disturbance area. Chapter 3, Section 3.1.5.3 identifies this area and Chapter 4, Section 4.5.1, discusses limitations for activities in the vicinity of the pygmy rabbit burrow.

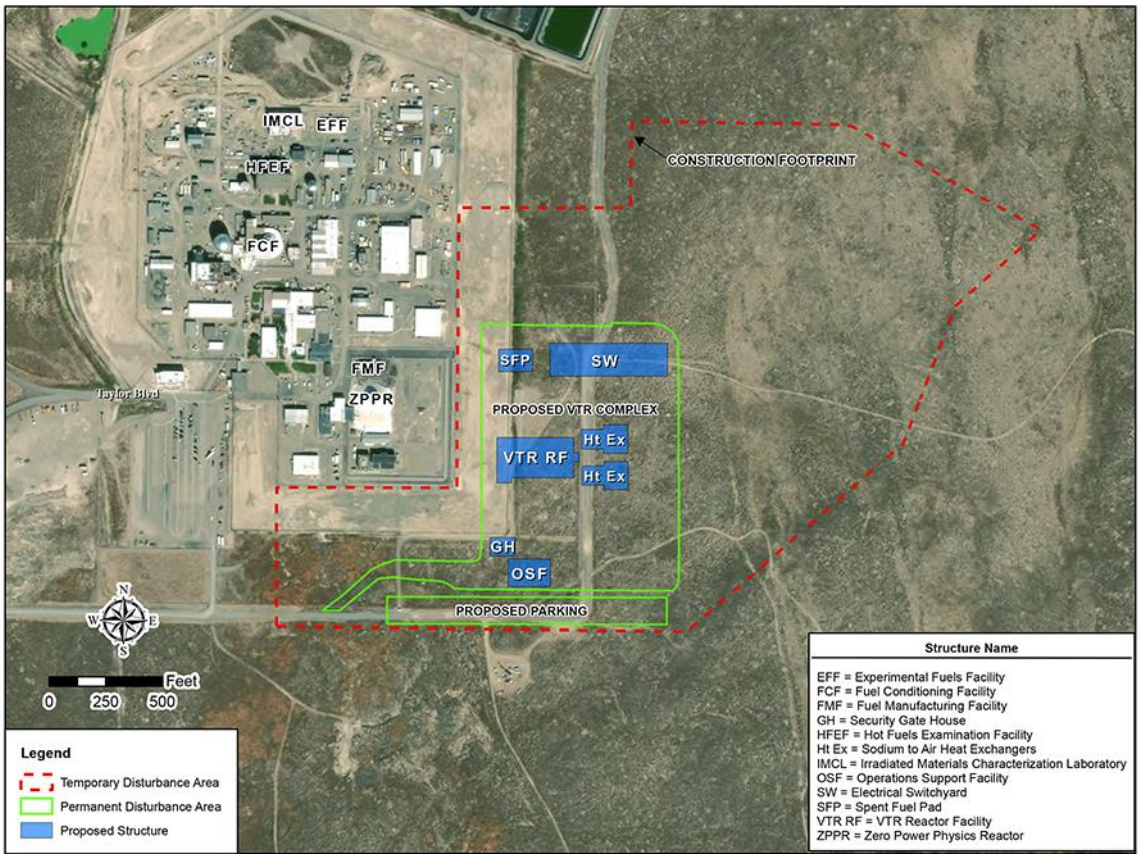


Figure B–15. Proposed Versatile Test Reactor Location at Idaho National Laboratory

VTR utility demands (electricity, water, etc.) would be supplied by existing MFC utility systems. With one exception, no modifications to the MFC utility systems would be required to support the addition of the VTR. The addition of the VTR to the MFC would require an upgrade to the electrical distribution system at the INL Site. A dynamic volt-ampere reactive device would be installed at the Advanced Test Reactor electrical substation to ensure electrical (voltage) stability for the area.

B.2.10.1 Environmental Resources – Construction

Resource Requirements

Table B–8 provides a summary of the key resources committed to the construction of the VTR facilities. The construction effort would ramp up until peaking in the third year of construction.

Table B–8. Idaho National Laboratory Resource Requirements During Versatile Test Reactor Construction

Resource	*Units*	*Annual Average Value*	*Annual Peak Value*	*Total* [a]
Staff	FTE	640	1300	2,700
Electricity	kWh	1,000,000	2,000,000	4,300,000
Gasoline	gallons	87,000	145,000	370,000
Diesel Fuel				
Road Diesel	gallons	84,000	144,000	360,000
Non-road Diesel	gallons	447,000	750,000	1,900,000
Total Diesel	gallons	531,000	894,000	2,300,000
Water				
Potable	gallons	8,000,000	16,000,000	34,000,000
Dust control, etc.	gallons	22,000,000	40,000,000	94,000,000
Total	gallons	30,000,000	56,000,000	128,000,000
Asphalt	cubic yards	---	---	1,400
Structural Concrete	cubic yards	---	---	40,000
Rebar	tons	---	---	4,350
Excavation	bank cubic yards [b]	---	---	135,000
Backfill Material	cubic yards	---	---	200,000 [c]
Landscaping	cubic yards	---	---	2,000
Structural Steel	tons	---	---	4,150
Large Bore Piping	linear feet	---	---	31,500
Cable and Wire	linear feet	---	---	1,200,000
Cable Tray	linear feet	---	---	18,000
Conduit Above Grade	linear feet	---	---	220,000
Conduit Inside Duct Banks	linear feet	---	---	53,000
Rock/Gravel	cubic yards	---	---	45,000
Temporary Concrete	cubic yards	---	---	14,000
Lumber	tons	---	---	250
Temporary Steel	tons	---	---	50
Gas [d]	bottles/cubic meters	---	---	20,000/130,000

FTE = full-time equivalent (person); kWh = kilowatt-hour.

[a] Construction duration of 51 months is assumed.

[b] A bank yard is the volume of earth or rock in its natural state, as compared to the expanded volume after excavation.

[c] Excavated material would be temporarily stored within the construction footprint and would be used as backfill. Material from a borrow site would be used for the additional 65,000 cubic yards needed.

[d] Gas bottles (cylinders) can range from 2 to 10 cubic meters in size. A typical size of 6.5 cubic meters has been used to estimate the volume of gas in the cylinders.

Source: INL 2020c.

Nonradiological Releases

Nonradiological releases are primarily associated with the operation of trucks and construction equipment (i.e., the burning of diesel fuel). However, fugitive dust contributes the majority of particulate matter emissions. Emission sources and air pollutant emissions are presented in **Table B–9**.

Table B–9. Calendar Year Nonradiological Construction Emissions – Idaho National Laboratory Versatile Test Reactor

Calendar Year/Source Type	*Air Pollutant Emissions (tons per year)*						
	VOC	*CO*	*NO_x*	*SO_2*	*PM_{10}*	*$PM_{2.5}$*	*CO_2e (metric tons)*
Year 2022							
Onsite On-road Sources	0.05	1.00	0.48	0.002	0.06	0.02	261
Onsite Nonroad Sources	0.35	2.47	4.66	0.01	0.27	0.27	1,614
Fugitive Dust	---	---	---	---	56.78	5.68	---
Offsite On-road Sources	0.08	5.12	1.00	0.006	0.20	0.05	761
Total Annual Emissions	**0.48**	**8.59**	**6.13**	**0.02**	**57.31**	**6.01**	**2,637**
Year 2023							
Onsite On-road Sources	0.08	1.46	0.78	0.004	0.09	0.04	445
Onsite Nonroad Sources	0.73	4.61	8.59	0.02	0.47	0.45	2,755
Fugitive Dust	---	---	---	---	102.21	10.22	---
Offsite On-road Sources	0.36	24.37	4.28	0.03	0.95	0.22	3,666
Total Annual Emissions	**1.16**	**30.44**	**13.64**	**0.05**	**103.72**	**10.93**	**6,866**
Year 2024							
Onsite On-road Sources	0.06	1.27	0.61	0.003	0.08	0.03	393
Onsite Nonroad Sources	0.68	4.16	8.50	0.02	0.43	0.41	2,773
Fugitive Dust	---	---	---	---	68.14	6.81	---
Offsite On-road Sources	0.32	24.32	3.91	0.03	0.98	0.22	3,763
Total Annual Emissions	**1.06**	**29.75**	**13.03**	**0.05**	**69.62**	**7.47**	**6,929**
Year 2025							
Onsite On-road Sources	0.02	0.73	0.22	0.002	0.04	0.01	182
Onsite Nonroad Sources	0.21	1.50	2.50	0.01	0.13	0.13	1,051
Fugitive Dust	---	---	---	---	34.07	3.41	---
Offsite On-road Sources	0.03	1.09	0.50	0.00	0.10	0.02	336
Total Annual Emissions	**0.26**	**3.32**	**3.21**	**0.01**	**34.33**	**3.57**	**1,569**

CO = carbon monoxide; CO_2e = carbon dioxide equivalent; NA = not applicable; NO_x = nitrogen oxides; $PM_{2.5}$ = particulate matter less than 2.5 microns in diameter; PM_{10} = particulate matter less than 10 microns in diameter; SO_2 = sulfur dioxide; VOC = volatile organic compound; --- = no air pollutant emission from this source type.
Source: Derived from INL 2020c.

Waste Generation

Table B–10 provides estimates of the wastes generated during VTR construction; this includes construction of all of the facilities (Reactor Facility, switchyard, exterior HRS components, the Operational Support Facility, and associated structures). There would not be any radiological waste generated during construction of the VTR.

Table B–10. Wastes Generated During Versatile Test Reactor Construction

Waste Type	*Material*	*Units*	*Value*
Hazardous Waste			Assumed to be 2 percent of nonhazardous waste volumes
Nonhazardous Waste	Concrete	cubic yards	9,900
	Rebar	tons	180
	Structural steel	tons	330
	Large bore pipe	linear feet	2,500
	Small bore pipe	linear feet	2,800
	Cable and wire	linear feet	96,000
	Cable tray	linear feet	1,400
	Conduit	linear feet	26,000
	Tubing	linear feet	2,800
	Instruments	each	65
	Valves	each	30
	In-line components	each	65
	Lumber	tons	120
	Steel	tons	50
	Gas bottles	bottles	19,200

Source: INL 2020c.

B.2.10.2 Environmental Resources – Operations

The nominal test cycle length for the VTR would be 100 effective full-power days. At the end of each cycle there would be a 20-day refueling cycle during which 14 to 15 driver fuel assemblies and test assemblies at the end of their planned test exposure times would be removed from the core (INL 2020c).

Resource Requirements

Key annual resource commitments for the operation of the VTR are provided in **Table B–11**. Annual staffing requirements include both the normal operational and maintenance staff for the VTR, as well as augmented staffing during refueling. Diesel fuel would be required for testing of the site diesel generators, and electric or propane heaters would be used as the heat source for the SAHX air pre-heaters. Since the VTR would be a sodium-cooled reactor, both the PHTS and HRS would use sodium coolant. The commitment of water would be required only for staff needs and firewater (system testing, etc.). No water would be used for cooling the reactor. Only chemicals used in quantities of over 1,000 pounds are shown in the table. Other chemicals would be used in smaller quantities (INL 2020d).

Table B–11. Annual Resource Requirements During Versatile Test Reactor Operation

Resource	*Units*	*Value*
		Annual (Peak)
Staff	FTE	200
Electricity [a]	MWh	140,000 (170,000)
Diesel Fuel [b]	gallons	9,200
Propane [c]	Standard cubic feet	18,500 (1,500,000)
Water		
Potable	gallons	1,200,000
Fire Water	gallons	1,700,000
Demineralized Water	gallons	250,000
Total	gallons	3,100,000
Chemicals		
Sulfuric Acid	pounds	640,000
Gasoline	pounds	79,000
Oil	pounds	59,000

Resource	Units	Value
		Annual (Peak)
Fuel Maintenance	pounds	20,000
Paint	pounds	10,000
Alcohol	pounds	13,000
Vehicle Maintenance	pounds	8,000
Adhesive	pounds	7,000
Cleaner	pounds	7,500
Building Maintenance	pounds	3,000
Lubricant	pounds	9,400
Sealant	pounds	2,500
Acetone	pounds	2,200
Grounds Keeping	pounds	1,900
Metal Cleaner	pounds	2,000
Coolant	pounds	1,400
Sodium Hypochlorite	pounds	1,200
Nitric Acid	pounds	6,400
Ammonium Hydrozide	pounds	7,000
Epoxy	pounds	3,400
Antifreeze	pounds	1,700
Caulk	pounds	1,300
Gases		
Compressed Neon	liters	23,000
Suva Refrigerant	pounds	5,200
Liquid Nitrogen	standard cubic feet	3,400
P-10 Gas (argon with 10% methane)	standard cubic feet	3,100
Methane	standard cubic feet	2,900
Freon (R-410a)	pounds	1,800
Hydrogen/Air Mix	liters	1,800
Compressed Helium	standard cubic feet	1,500
Compressed Oxygen	standard cubic feet	1,200

FTE = full-time equivalent (person); MWh= megawatt-hours.

[a] Annual electricity usage was provided in MVa (mega-volt-amperes). A load factor of .9 was used to convert to MWs (megawatts).

[b] Diesel generators would operate 1 percent of the time, 88 hours per year. Fuel consumption is based on the fuel consumption rates (Leidos 2020).

[c] Propane heaters are an alternative design for preheating air in the sodium-to-air heat exchangers. Use of this alternative design would be a site-specific decision. These heaters would be used for short periods when the reactor is shutdown following a test cycle. The peak usage is associated with an extended maintenance outage, projected to be needed once every 15 years.

Source: GE Hitachi 2019b; INL 2020c.

Nonradiological Releases

The main source of nonradiological releases associated with the operation of the VTR would be the releases from operation of the site diesel generators, personal vehicles, and vehicles used to transport materials (wastes, spent fuel, test assemblies, etc.). The generators supply power to the site in the event of a loss of the normal offsite power supply. To ensure that the generators are functional, they would be tested, started and run for a period of time, several times a year. The annual emissions associated with these sources are provided in **Table B–12**.

Table B–12. Versatile Test Reactor Operational Nonradiological Emissions

Emission Source	Air Pollutant Emissions (tons per year)							
	VOC	CO	NO_x	SO_2	PM_{10}	$PM_{2.5}$	CO_2	CO_2e (metric tons)
Back-up Generators – VTR	0.03	0.50	0.10	0.00	0.00	0.00	102	93
Pre-Heaters – Normal Annual	0.00	0.00	0.00	0.00	0.00	0.00	3	3
Haul Trucks	0.03	0.15	0.55	0.00	0.08	0.02	305	277
Worker Commuter Vehicles	0.02	2.85	0.18	0.00	0.08	0.02	382	347
Total – Normal Annual Operations	**0.09**	**3.50**	**0.84**	**0.01**	**0.17**	**0.04**	**793**	**720**
Pre-Heaters –Large Component Replacement [a]	0.02	0.16	0.27	0.00	0.01	0.01	263	239
Total Annual Emissions [b]	**0.11**	**3.66**	**1.11**	**0.01**	**0.18**	**0.06**	**1,052**	**956**

CO = carbon monoxide; CO_2e = carbon dioxide equivalent; NO_x = nitrogen oxides; $PM_{2.5}$ = particulate matter less than 2.5 microns in diameter; PM_{10} = particulate matter less than 10 microns in diameter; SO_2 = sulfur dioxide; VOC = volatile organic compound.

[a] Large Component Replacement would occur every 15 years.

[b] Equal to sum of Back-up Generators, Haul Trucks, Worker Commuter Vehicles, and Pre-Heaters Large Component Replacement.

Source: Derived from INL 2020d.

Radiological Releases

Radiological releases were estimated assuming that the VTR operates for three test cycles per year of 100 days each, with one failed fuel pin in the core at all times. The estimated annual release activity per isotope is presented in **Table B–13**.

Table B–13. Versatile Test Reactor Operational Annual Radiological Releases

Isotope	Annual Release (curies)	Isotope	Annual Release (curies)
Argon-41 [a]	27.1	Krypton-88	8.9×10^{-06}
Cesium-135	9.0×10^{-16}	Xenon-131m	1.6×10^{-02}
Cesium-137	1.2×10^{-12}	Xenon-133	1.0×10^{-03}
Cesium-138	2.0×10^{-06}	Xenon-133m	5.4×10^{-07}
Hydrogen-3 (Tritium)	1.2	Xenon-135	4.2×10^{-05}
Krypton-83m	1.8×10^{-06}	Xenon-135m	1.5×10^{-06}
Krypton-85	0.70	Xenon-137	7.4×10^{-07}
Krypton-85m	3.5×10^{-06}	Xenon-138	4.4×10^{-06}
Krypton-87	4.8×10^{-06}		

[a] Most of the release of argon (27 curies) is through the RVACS stacks. The rest (0.01 curies) is through the facility HVAC stacks.

Source: INL 2020c.

Note that currently the only anticipated normal operation releases of radioactivity to the environment, with the exception of most of the argon, would be from the Gaseous Radioactive Waste System. The release from the Gaseous Radioactive Waste System would be inserted into the radioactive waste area HVAC system exhaust. The combined flow rate would be about 2,400 cubic meters per minute, at approximately 105 °F. The HEPA-filtered release would be through a 24-inch diameter stack, at a height of about 99 feet. The HVAC systems, Liquid Radioactive Waste System, and Solid Radioactive Waste System are not anticipated to have appreciable releases to the environment. The unfiltered releases of argon from activated air would be from the RVACS stacks would be from four 7-foot diameter RVACS

stacks, outer diameter, at an elevation of approximately 98 feet, with a total flow rate of 1,000 cubic meters per minute, at a temperature less than 500 °F (INL 2020c).

Waste Generation

Annual waste generation rates, based on three test cycles per year, are presented in **Table B–14**.

Table B–14. Versatile Test Reactor Operational Annual Waste Generation

Waste Type	*Category*	*Annual Average Volume (cubic meters)*		*Average Weight Maximum*	
		Net	*Gross*	*Net*	*Gross*
Hazardous waste	NA	3.2	4.4	5,400	6,500
Industrial	NA	22	26	27,000	30,000
Universal	NA	0.88	0.99	420	490
TSCA	NA	2.3	2.4	1,300	1,900
Recyclable	NA	4.5	6.0	9,700	11,000
Low-level waste	Contact handled	160	180	58,000	98,000
Mixed low-level waste	Contact handled	4.7	5.9	7,000	8,800
	Remote handled	0.7	1.7	280	4,700
Waste Type	**Unit**	**Quantity**			
Driver fuel assemblies	assemblies	45/66 [a]			
Liquid low-level waste	gallons	250,000			
Sanitary waste	gallons	1,200,000			

NA = not applicable; TSCA = Toxic Substance Control Act material.

[a] Up to 45 assemblies could be removed during a single year consisting of three operational cycles. Sixty-six assemblies would be removed from the VTR when the final core is removed.

Source: INL 2017b, 2020d; GE Hitachi 2019b.

B.2.11 Versatile Test Reactor at Oak Ridge National Laboratory

At Oak Ridge National Laboratory (ORNL), the VTR would be built approximately a mile east of the High Flux Isotope Reactor complex. Construction of the VTR has been estimated to take approximately 51 months, once design activities are complete. Based on the layout of the VTR (see Section B.2.2), the VTR complex at ORNL would occupy about 25 acres. However, in addition to the construction of the VTR, additional test assembly examination and spent fuel treatment and storage facilities would be constructed at ORNL (see Sections B.3.4 and B.4.4). These facilities would be collocated with the VTR, and in total, would result in a land commitment to the VTR and facilities of less than 50 acres. The test assembly examination and spent fuel treatment facility, spent fuel pad, and most of the VTR structures would be enclosed in a PIDAS. During construction, an additional 100 acres would be required for temporary parking and equipment laydown, assembly, and staging. In total, up to 150 acres would be impacted by the VTR, test assembly examination facility, and fuel storage pad construction (see **Figure B–16**).

VTR utility demands (electricity, water, etc.) would be supplied by existing ORNL utility systems. Once connected, no modifications to the ORNL utility systems would be required to support the addition of the VTR.

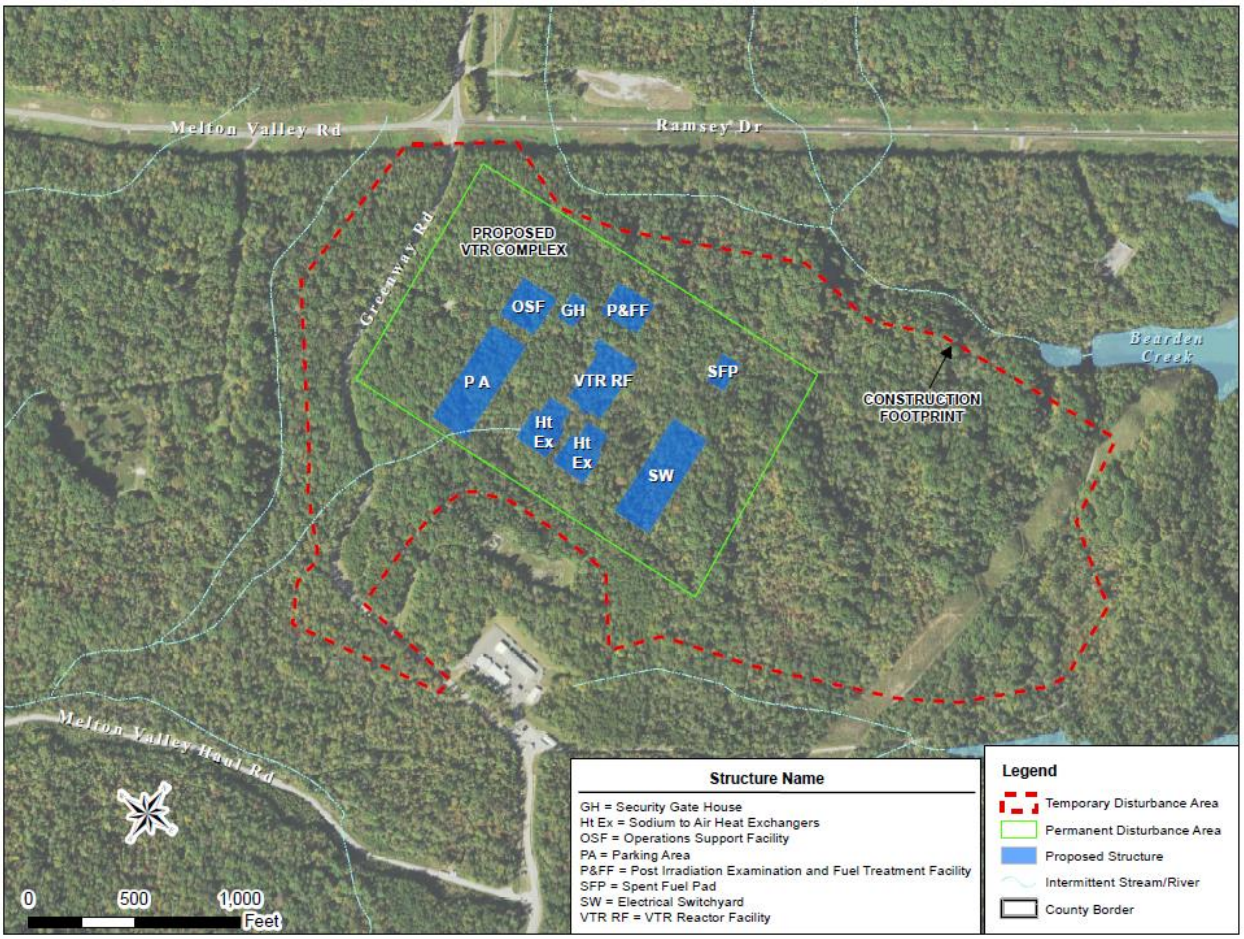


Figure B–16. Proposed Versatile Test Reactor Location at Oak Ridge National Laboratory

B.2.11.1 Environmental Resources – Construction

Resource Requirements

The environmental resources required or affected by construction of the VTR at ORNL would be similar to those described for the INL Site in Section B.2.10.1, but would include resources required for site preparation of an undisturbed, wooded area. Unlike at the INL Site, trees would need to be removed and the site more extensively graded. Resource requirements for site preparation at ORNL are presented in **Table B–15**. Once the site is prepared, resources required for the construction of the VTR facilities (VTR Reactor Facility, switchyard, sodium-to-air heat exchangers, Operational Support Facility, etc.) would be the same as those presented for VTR construction at INL (see Table B–8), with two exceptions. Construction at ORNL would involve the construction of a shorter road from existing roads to the facility parking lot, this results in a reduction in the use of asphalt (about 400 cubic yards less).[11] The construction activities at ORNL would include construction of the test assembly examination and spent fuel treatment facility. The resources affected by construction of this facility are discussed in Sections B.3.4 and B.4.4, respectively.

[11] Differences between INL and ORNL access road and parking lot construction resource utilizations for other resources are small and do not change the values presented in Table B–8.

Table B–15. Resource Requirements During Versatile Test Reactor Site Preparation at Oak Ridge National Laboratory

Site Preparation				
Resource	***Units***	***Annual Average Value***	***Annual Peak Value***	***Total*** [a]
Staff	FTE	16	NA	16
Diesel Fuel				
Road Diesel	gallons	25,000	NA	25,000
Non-road Diesel	gallons	244,000	NA	244,000
Total Diesel	gallons	269,000	NA	269,000
Gasoline	gallons	300	NA	300
Water				
Potable Water	gallons	250,000	NA	250,000
Dust Control	gallons	140,000	NA	140,000
Total Water	gallons	390,000	NA	390,000
Excavation [b]	cubic yards	690,000	NA	690,000
Fill Material	cubic yards	29,000	NA	720,000

FTE = full-time equivalent (person); NA = not applicable.

[a] Site preparation duration of about 10 months; includes 5 months for tree removal and 5 months for site grading.

[b] Excavated material would be temporarily stored within the construction footprint and would be used as backfill. Material from a borrow site would be used for the additional 29,000 cubic yards needed.

Source: Leidos 2020.

Nonradiological Releases

Nonradiological releases are associated with the operation of trucks and construction equipment (i.e., the burning of diesel fuel). Types and duration of operation for the equipment used during construction are discussed in the main body of this EIS. For construction of the VTR at ORNL, the nonradiological emissions would include those associated with site preparation as well as facility construction, and are presented in **Table B–16**.

Table B–16. Oak Ridge National Laboratory Site Preparation and Facility Construction Nonradiological Emissions

Year/Activity-Source Type	***Emissions (tons)***							***Combined HAPs*** [a]	***CO_2e (mt)***
	VOC	***CO***	***NOx***	***SO_2***	***PM_{10}***	***$PM_{2.5}$***	***CO_2***		
Year 2022									
Onsite Emissions from On-road Sources	0.01	0.14	0.15	0.00	0.02	0.01	79	0.00	72
Onsite Emissions from Nonroad Sources	0.33	1.80	0.99	0.00	0.08	0.08	300	0.05	272
Fugitive Dust	---	---	---	---	6.95	0.69	---	---	---
Offsite Emissions from On-road Sources	0.03	0.32	0.54	0.00	0.07	0.02	260	0.01	236
Slash Burning	28.88	136.80	3.06	1.91	26.64	22.64	3,065	1.09	2,787
Total 2022 Emissions	**29.26**	**139.06**	**4.75**	**1.91**	**33.76**	**23.45**	**3,704**	**1.15**	**3,367**
Year 2023									
Onsite Emissions from On-road Sources	0.08	2.99	0.62	0.00	0.10	0.03	607	0.02	552
Onsite Emissions from Nonroad Sources	0.68	4.27	8.45	0.02	0.42	0.41	2,828	0.11	2,571
Fugitive Dust	---	---	---	---	43.21	4.32	---	---	---
Offsite Emissions from On-road Sources	0.11	6.93	1.20	0.01	0.27	0.06	1,161	0.02	1,055
Total 2023 Emissions	**0.87**	**14.19**	**10.27**	**0.03**	**44.02**	**4.82**	**4,596**	**0.15**	**4,178**

Year/Activity-Source Type	Emissions (tons)							Combined HAPs [a]	CO_2e (mt)
	VOC	CO	NOx	SO_2	PM_{10}	$PM_{2.5}$	CO_2		
Year 2024									
Onsite Emissions from On-road Sources	0.18	8.06	1.46	0.01	0.28	0.08	1,621	0.04	1,474
Onsite Emissions from Nonroad Sources	1.06	6.21	12.75	0.03	0.62	0.60	4,031	0.18	3,665
Fugitive Dust	---	---	---	---	13.35	1.33	---	---	---
Offsite Emissions from On-road Sources	0.29	19.68	3.31	0.03	0.83	0.18	3,478	0.07	3,162
Total 2024 Emissions	**1.53**	**33.96**	**17.52**	**0.07**	**15.08**	**2.19**	**9,131**	**0.28**	**8,301**
Year 2025									
Onsite Emissions from On-road Sources	0.13	5.55	1.14	0.01	0.22	0.06	1,273	0.03	1,157
Onsite Emissions from Nonroad Sources	1.00	5.64	12.35	0.03	0.58	0.56	4,303	0.17	3,912
Fugitive Dust	---	---	---	---	7.32	1.08	0	---	---
Offsite Emissions from On-road Sources	0.18	13.45	2.15	0.02	0.60	0.12	2,483	0.04	2,257
Total 2025 Emissions	**1.31**	**24.64**	**15.64**	**0.06**	**8.72**	**1.83**	**8,058**	**0.24**	**7,326**

CO = carbon monoxide; CO_2 = carbon dioxide; CO_2e = carbon dioxide equivalent; HAPs = hazardous air pollutants; mt = metric tons; NOx = nitrogen oxides; $PM_{2.5}$ = particulate matter less than 2.5 microns in diameter; PM_{10} = particulate matter less than 10 microns in diameter; SO_2 = sulfur dioxide; VOC = volatile organic compound; --- = no pollutant emissions from this source type.

[a] Combined HAPs = 15/3 percent of combustive VOC/PM emissions for on-road and nonroad sources and 1/3 percent for slash burning (California Air Resources Board 2018).

Source: Derived from Leidos 2020.

Waste Generation

Estimates of the wastes generated during VTR construction at ORNL would be the same at ORNL as at INL (see Table B–10), this includes waste from the construction of the reactor facilities (Reactor Facility, switchyard, exterior HRS components, the Operational Support Facility, and associated structures). There would not be any radiological waste generated during construction of the VTR. Marketable material from the trees removed during site preparation would be shipped to a local lumberyard, the remainder mulched or burned onsite. Excavation material would be used onsite for site backfill. Therefore, the site preparation activities would not result in the generation of any waste requiring disposal.

B.2.11.2 Environmental Resources – Operations

Resource Requirements

The environmental resources required for operation of the VTR at ORNL would be the same as those described for INL in Section B.2.10.2.

Nonradiological Releases

The main source of nonradiological releases associated with the operation of the VTR would be the releases from operation of the site diesel generators, operations staff personal vehicles, and vehicles used to transport materials (wastes, spent fuel, test assemblies, etc.). The generators supply power to the site in the event of a loss of the normal offsite power supply. To ensure that the generators are functional, they would be tested, started, and run for a period of time, several times a year. The annual emissions associated with these generators are presented in **Table B–17**. Emissions presented in this table include those for all activities at the VTR site, including VTR reactor operations, post-irradiation examination, and spent fuel treatment and storage.

Table B–17. Versatile Test Reactor Operational Nonradiological Emissions

Emission Source	Air Pollutant Emissions (tons per year)							CO_2e (metric tons)
	VOC	CO	NO_x	SO_2	PM_{10}	$PM_{2.5}$	CO_2	
Back-up Generators – VTR	0.05	0.66	0.13	0.001	0.01	0.01	133	121
Pre-Heaters – Normal Annual	0.00	0.00	0.00	0.00	0.00	0.00	3	3
Haul Trucks	0.03	0.13	0.45	0.00	0.07	0.02	271	246
Worker Commuter Vehicles	0.04	4.81	0.26	0.00	0.13	0.02	624	568
Total –Normal Annual Operations	**0.11**	**5.60**	**0.84**	**0.01**	**0.20**	**0.05**	**1,031**	**938**
Pre-Heaters –Large Component Replacement [a]	0.02	0.16	0.27	0.00	0.01	0.01	263	239
Total Annual Emissions [b]	**0.13**	**5.75**	**1.11**	**0.01**	**0.22**	**0.06**	**1,291**	**1,173**

CO = carbon monoxide; CO_2 = carbon dioxide; CO_2e = carbon dioxide equivalent; HAPs = hazardous air pollutants; NO_x = nitrogen oxides; PM_{10} = particulate matter less than 10 microns in diameter; $PM_{2.5}$ = particulate matter less than 2.5 microns in diameter; SO_2 = sulfur dioxide; VOC = volatile organic compound; VTR = Versatile Test Reactor.

[a] Large Component Replacement would occur every 15 years.

[b] Equal to sum of Back-up Generators, Haul Trucks, Worker Commuter Vehicles, and Pre-Heaters Large Component Replacement.

Source: Derived from Leidos 2020.

Radiological Releases

The radiological releases from operation of the VTR at ORNL would be the same as those described for INL in Section B.2.10.2 and presented in Table B–13.

Waste Generation

The waste generated from operation of the VTR at ORNL would be the same as those described for INL in Section B.2.10.2 and presented in Table B–14.

B.3 Test Assembly Examination

B.3.1 Introduction

Test assemblies from the VTR would be temporarily stored in the VTR Reactor Facility, within the reactor vessel, if necessary, to allow the assembly to cool sufficiently for handling and transport. Some prompt post-irradiation examination of a test assembly may be performed in a shielded cell located in a pit at the VTR Reactor Facility. Most post-irradiation examination would occur at separate facilities collocated with the VTR.

B.3.2 Post-Irradiation Examination of Test Assemblies

Concurrent with the irradiation capabilities provided by the VTR, the mission need requires the capabilities to examine the test specimens irradiated in the reactor to determine the effects of a high flux of high-energy or fast neutrons. The test specimens could include assemblies of fuel or materials often encapsulated in cartridges such that the material being tested is fully contained. The highly radioactive test specimen capsule would be removed from the reactor after a period of irradiation, ranging from days to years, depending on the nature of the test requirements, and transferred to a fully shielded facility where the test item could be analyzed and evaluated remotely. The examination facilities are “hot-cell” facilities (see **Figure B–17**). These hot cells include concrete walls several feet thick; multi-layered, leaded-glass windows several feet thick; and remote manipulators that allow operators to perform a range of tasks remotely without incurring a substantial radiation dose from the test specimens within the hot cell. In some cases, an inert atmosphere is required to prevent test specimen degradation. DOE intends that the hot-cell facilities where the test items are examined and analyzed after removal from the reactor, would be in close proximity to the VTR to minimize onsite or offsite transportation of the potentially high-radioactive specimens.

Figure B–17. Exterior and Interior Views of Hot Cell Facilities

Needed testing capabilities would include the ability to assess macro and microscopic changes to irradiated materials. Irradiated materials (test specimens) could include reactor fuels, coolants, and any other material that could be exposed to a fast flux in a demonstration or operating fast reactor (e.g., any liquid metal cooled, molten salt fueled and cooled, gas cooled). The post-irradiation examination facility must have the ability to disassemble the test assemblies and the test specimens (disassembly of a test capsule if used and the test specimen itself) and should be able to perform non-destruction examination of irradiated samples including dimensional measurements and neutron radiography (NRAD), and destructive examination including mechanical testing or microscopic examination and characterization of metals and/or ceramics.

B.3.3 Test Assembly Examination at the Idaho National Laboratory Site

B.3.3.1 Facilities

Test assembly examination at INL could be performed at the Hot Fuel Examination Facility (HFEF), the Irradiated Materials Characterization Laboratory (IMCL), the Analytical Laboratory, and the Electron Microscopy Laboratory. Test assemblies would first be transferred to the HFEF for initial disassembly and examination. Entire test specimens or portions of specimens could be transferred to the other facilities to make use of their specialized examination capabilities. The existing facilities would not require modification; although, the HFEF would need new in-cell handling equipment for experiment movements (INL 2020c). All facilities currently do test assembly examination and are able to accept casks with radioactive material. The HFEF can currently accept the test assemblies and dismantle the assemblies for shipment to other facilities (INL 2020c). The facility is linked to analytical laboratories and other facilities by pneumatic sample transfer lines (INL 2017a).

The HFEF, the largest hot-cell facility at INL, is a versatile hot cell facility that consists primarily of two adjacent shielded cells, the main cell and the decontamination cell, surrounded by offices, laboratories, and personnel-related areas in a three-story (above-ground) building. A service level is located below ground. The facility includes an air-atmosphere decontamination cell, an argon-atmosphere main cell (the main cell), decontamination areas, repair areas for hot-cell equipment, auxiliary laboratories, offices, and a high bay area (INL 2020c).

The main cell is a 70 by 30 foot stainless steel-lined gas-tight hot cell. It is fitted with two 5-ton cranes and two electromechanical manipulators. There are 15 workstations, each with a 4-foot-thick window of oil-filled, cerium-stabilized high-density leaded glass and a pair of remote manipulators for use in its purified argon atmosphere. The decontamination hot cell is an air cell that includes five workstations and a water wash spray chamber for decontaminating materials and equipment. Assemblies would be dismantled using the precision mill, a low-speed mill (INL 2017a).

Non-destructive and destructive radioactive material examination and processing is performed in the decontamination cell and main cell. The radioactive materials involved in these activities include actinides and fission products. Radioactive material examination tasks include, but are not limited to, investigation of material characteristics (microstructure) and measurement of properties (fuel length, bowing, cladding surface distortion, and radionuclide distribution). Investigations of these phenomena are performed on samples ranging in mass from milligrams to hundreds of grams. The samples may be cut, ground, and/or polished to facilitate examination (INL 2020c).

These activities utilize current capabilities housed in the HFEF, including:

- gamma scanning,
- visual examination and eddy current testing,
- gas sampling using the Gas Assay Sample and Recharge,
- accident simulation testing in the Fuel and Accident Condition furnace,
- metallic and ceramic sample preparation, and
- bench measurements.

The HFEF also houses the NRAD reactor (a 300-kilowatt TRIGA [Training, Research, Isotopes, General Atomics] reactor), located in the HFEF basement. NRAD is a neutron source for radiographs of experiment components (INL 2017a).

Radioactive material is stored in the HFEF in various storage arrangements in the main cell and consists of (1) FFTF fuel; (2) EBR-II fuel in element magazines; and (3) uranium, plutonium, and other radioactive fuels or materials in containers of various shapes and sizes (INL 2020c).

The IMCL is a 12,000-square foot research facility and is the newest of the INL MFC facilities. The IMCL focuses on microstructural and thermal characterization of irradiated nuclear fuels and materials. The IMCL's design provides customizable radiological shielding and confinement systems. The shielded instruments allow characterization of highly radioactive fuels and materials at the micro-scale and nanoscale. The IMCL was designed to facilitate evolving capabilities (i.e., its flexible modular design would simplify the adaptation of its capabilities to support VTR nuclear fuel and materials examinations). The IMCL has free space for user-defined capability, such as the VTR program. Current and future planned capabilities include:

- Preparation of minute samples for further testing,
- Precision quantitative composition analysis,
- Microstructural characterization, and
- Thermal property measurement (INL 2019c).

In addition to the HFEF and IMCL, some post-irradiation examination could occur at the Analytical Laboratory and the Electron Microscopy Laboratory. The radiochemistry laboratory has six hot cells and eight gloveboxes and general chemistry laboratories. It has the capability to examine irradiated samples including fuels. Equipment within the laboratory can be used to test fundamental physical properties of samples and includes mass spectrometers and gamma and alpha counters (INL 2020a). The Electron Microscopy Laboratory performs materials characterization using electron and optical microscopy tools.

B.3.3.2 Environmental Resources – Construction

Test assembly inspection is currently performed in existing INL facilities. Significant modification of existing facilities is not anticipated. Modifications would consist of removal of some existing legacy equipment and replacement with new equipment that meets the VTR needs. This is a routine activity that is currently performed in these facilities. Any changes to resource requirements would be minimal (e.g.,

minimal water usage associated with manufacturing of tooling for equipment replacement). No additional plant staff would be required during construction and any changes to resource requirements would be minimal (INL 2020c).

B.3.3.3 Environmental Resources – Operations

The nominal test cycle length for the VTR would be 100 effective full-power days. At the end of each cycle, test assemblies at the end of their test exposure times would be removed from the core. The test specimens within the assemblies would be allowed to cool within the reactor vessel for a period of time. When removed from the reactor vessel and after being cleaned (sodium washed), these test assemblies would be transferred to the post-irradiation examination facility.

Resource Requirements

Most VTR-associated activities would be encompassed by the scope of current activities. No additional staff would be required, assuming that the VTR test specimen preparation and examination activities would supplant current activities at the HFEF. Resource requirements for VTR-related activities are presented in **Table B–18**. Only chemicals used in quantities of over 1,000 pounds are shown in the table. Other chemicals and gases would be used in smaller quantities (INL 2020d).

Table B–18. Idaho National Laboratory Annual Test Assembly Facility Operational Resource Requirements

Resource	*Units*	*Value Annual*
Staff	FTE	80 [a]
Electricity	MWh	minimal
Water		
Potable – staff	gallons	1,000,000
Component wipedown	gallons	1,000
Total	gallons	1,000,000
Chemicals		
Nitric Acid	pounds	17,000
Alcohol	pounds	9,300
Lubricant	pounds	1,400
Acetone	pounds	1,200
Hydrochloric acid	pounds	1,000
Gases		
Argon liquid	standard cubic feet	61,000
Argon/carbon dioxide/hydrogen/methane/methanol	liters	7,800

FTE = full-time equivalent (person); MWh = megawatt-hour.

[a] These are all existing staff members (Nelson 2020). VTR activities would replace existing activities.

Source: INL 2020c.

Nonradiological Releases

The nonradiological releases from the HFEF are not expected to change with the addition of VTR test assembly operations. No new sources of emissions are anticipated (INL 2020c).

Radiological Releases

Radiological releases were estimated to increase by 40 percent over current post-irradiation examination operations due to VTR-related activities. The estimated annual release activity per isotope is presented in **Table B–19**. The isotopes in bold are those that contributed at least 0.1 percent of the total offsite dose from MFC operations in 2018, based on the INL Annual Site Environmental Report (INL 2019d). Other isotopes listed are limited to those with releases greater than 10^{-10} curies.

Table B–19. Idaho National Laboratory Test Assembly Examination Facility Operational Annual Radiological Releases

Isotope	*Release (curies)*	*Isotope*	*Release (curies)*
Antimony-125	3.2×10^{-5}	Krypton-85	4.4×10^{-3}
Americium-241	$\mathbf{8.4 \times 10^{-12}}$	Neptunium-237	3.2×10^{-9}
Carbon-14	$\mathbf{3.1 \times 10^{-4}}$	Phosphorus-32	2.6×10^{-5}
Cadmium-109	5.2×10^{-4}	Phosphorus-33	4.9×10^{-9}
Cadmium-115m	1.0×10^{-7}	**Plutonium-238**	$\mathbf{1.2 \times 10^{-10}}$
Chlorine-36	1.0×10^{-5}	**Plutonium-239**	$\mathbf{9.5 \times 10^{-8}}$
Cobalt-60	$\mathbf{7.9 \times 10^{-13}}$	**Plutonium-240**	$\mathbf{3.0 \times 10^{-12}}$
Cesium-134	8.0×10^{-7}	Plutonium-242	1.8×10^{-9}
Cesium-137	$\mathbf{2.5 \times 10^{-2}}$	Sodium-22	3.2×10^{-6}
Hydrogen-3 (Tritium)	$\mathbf{3.7 \times 10^{-2}}$	Sodium-24	1.7×10^{-8}
Iodine-129	$\mathbf{1.8 \times 10^{-5}}$	Sulfur-35	1.2×10^{-4}
Iodine-131	8.9×10^{-3}	**Strontium-90**	$\mathbf{3.8 \times 10^{-7}}$

Note: The isotopes in bold are those that contributed at least 0.1 percent of the total offsite dose from MFC operations in 2018, based on the INL Annual Site Environmental Report (INL 2019d).
Source: INL 2020d.

Releases of radioactivity to the environment would be through the existing release points for each of the facilities that could be used for post-irradiation examination. All test specimens would be processed through the HFEF first; individual samples could be transferred to other facilities for detailed examination. The combined flow rate would be about 35,200 cubic feet per minute at 72 °F. The release would be through a rectangular, 84 by 30-inch stack, at a height of about 95 feet.

Waste Generation

Waste from post-irradiation examination activities would involve discarding of material from driver fuel assemblies and experiments as well as low-level waste items associated with cask operations and operator protective equipment (INL 2020c). Annual waste generation rates, based on the handling of up to 60 test assemblies per year, are provided in **Table B–20**.

Table B–20. Idaho National Laboratory Test Assembly Facility Annual Waste Generation

Waste Type	*Category*	*Volume (cubic meters)*		*Weight (pounds)*	
		Net	*Gross*	*Net*	*Gross*
Hazardous	NA	1.6	4.7	1,400	2,300
Industrial	NA	1.9	1.9	1,300	1,600
Recyclable	NA	1.2	1.2	1,900	2,000
TSCA	NA	0.053	0.054	70	87
Universal	NA	0.12	0.13	83	95
Low-level waste	Contact handled	93	100	35,000	50,000
	Remote handled	2.5	2.6	1,900	2,800
Mixed low-level waste	Contact handled	6.3	8.9	7,800	9,800
Transuranic waste	Contact handled	0.67	0.75	310	540
Mixed transuranic waste	Contact handled	0.14	0.14	62	100
	Remote handled	0.073	0.11	90	470

NA = not applicable.
Source: INL 2020d.

B.3.4 Test Assembly Examination at Oak Ridge National Laboratory

B.3.4.1 Facilities

Test assembly post-irradiation examination at ORNL would make use of some existing facilities, but none of these facilities would include hot cells that operate using an inert environment; all would use an air atmosphere. Initial test assembly examination activities would need to be performed within a hot cell with an inert atmosphere. Once properly prepared, additional examination of the test specimens can be performed at existing ORNL facilities.

A new hot cell facility with inert atmosphere hot cells adjacent to the VTR would be needed. A conceptual design[12] for this facility has been developed to meet the process requirements identified in Section B.3.2, using equipment similar to that identified under the INL Alternative for the VTR (Section B.3.3). The facility would be located adjacent to the VTR, within a common protected area, and would support both test specimen post-irradiation examination and spent fuel treatment activities. In size and capability, this new post-irradiation examination facility would be similar to the INL HFEF (see Section B.3.3.1).

New Facility

The new hot cell facility would provide an inerted hot cell for post-irradiation examination (plus one for spent fuel treatment, See Section B.4.4). Each hot cell would be connected to a decontamination cell with an air atmosphere. The hot cell facility would have four levels and would be approximately rectangular with a reinforced concrete structure. The bottom portion of the hot cell facility would have a footprint of about 172 by 154 feet.

The hot cell facility would include two major structural systems: a concrete structure from the basement level up to the floor of the fourth level or high bay area, and a steel structure enclosing the fourth level high bay area.

The reinforced concrete bottom portion of the hot cell facility would consist of three floors: the service floor, an operating floor, and a second floor extending from an elevation of about -16 feet (16 feet below surface level) to the top of the second floor at 29 feet. The concrete structure would contain the test assembly hot cell, the spent fuel treatment hot cell, and the two associated decontamination cells. The top of the concrete structure forms the floor of the high bay area.

A steel-braced structure, 122 by 154 feet, would rise about 53 feet above the concrete portion of the structure. This high bay area would be constructed of metal siding and a metal roof deck, at an elevation of about 86 feet above ground level, supported by steel roof beams and tapered, built-up, steel roof girders. The steel structure would form a high bay for a 40-ton overhead crane, used to transfer equipment and material, including transfer of material between the truck lock, high bay, and cask tunnel.

A hot repair area would be an enclosed single-story area near the center of the high bay area. This area would be used for the maintenance of in-cell material-handling equipment. The area would not be shielded, as equipment would be decontaminated prior to being moved to this area. The hot repair area would be constructed of concrete-block masonry perimeter and interior walls, with a roof of steel decking covered by a thin layer of concrete.

The new hot cell facility would include a truck lock to accommodate receipt of the various materials into the facility through roll-up doors at each end. A 25-ton bridge crane in the top of the truck lock would be provided to move loads through a floor hatch into the cask tunnel for each hot cell. The ceiling of the truck lock would consist of metal covers that could be removed for access to the high bay area. A 29.5-foot-deep cylindrical cask handling pit is included. The truck lock would also be accessible to the 40-ton

[12] The conceptual designs have been developed for National Environmental Policy Act purposes only. This conceptual design is not as detailed as, nor is it to be considered, the conceptual design that is a part of the DOE facility design process.

HBA crane, which would be used to move loads between the truck lock, the high bay area, and the cask tunnel.

Transfer tunnels would be incorporated into the hot cell facility design, a cask tunnel and shielded transfer tunnels. The cask tunnel would be used to transfer material (equipment, tools, experiments, etc.) from top-opening casks into the cell complex. The cask tunnel would extend from the truck lock to the decontamination cells. The shielded transfer tunnels, located under the cell floors, would be used for the movement of large equipment and irradiated components between the decontamination cells and the inerted cells.

A central portion of the hot cell facility, measuring approximately 100 by 105 feet, would house the test assembly examination and spent fuel treatment hot cells, cask tunnel, and other facilities. These areas would have concrete walls and concrete-floor slabs for radiation-shielding purposes. The area surrounding this central cell area would house offices, labs, corridors, and other rooms. The floors in the office areas would be thin, reinforced concrete slabs, supported by reinforced concrete girder-joist systems, which, in turn, would be supported on reinforced concrete columns. The perimeter wall up to a grade elevation, would be constructed of reinforced concrete. Above this elevation, the walls and interior partitions would be concrete masonry blocks.

The test assembly examination portion of the hot cell facility would have its own set of inerted hot and decontamination cells. The test assembly examination hot cell would be a concrete-shielded, steel-lined enclosure with interior dimensions of 30 feet wide by 70 feet long by 25 feet high. It would be filled with argon gas that provides an inert, non-oxidizing atmosphere. The associated decontamination cell would be a concrete-shielded, steel-lined enclosure with interior dimensions of 30 feet wide by 20 feet long by 25 feet high; it would be filled with air. The interior surfaces of the cell would be lined with steel. A raised steel floor would extend over part of the cell. Sections of the raised floor could be removed for access to the subfloor area. Test samples and equipment would be moved using two 5-ton cranes and electromechanical manipulators. The space beneath the removable floor would be used for storage; it would also house gas ducts and filters, and serve as additional space (depth) for vertical handling of long items.

There would be penetrations in the cell walls, roof, and floor for windows, utility service, feedthroughs, in-cell handling equipment, gas ducting, transfer hatches, etc. Penetrations into each cell would be steel-lined, welded to the cell liner, and surrounded by high-density shielding closures or inserts. Closures or inserts for the penetration liners would have double seals, with the space between them pressurized with an argon purge.

The test assembly examination hot cell would have 15 work stations, each about 10 feet wide, equipped with a shielding observation window (layers of leaded glass with thin layers of mineral oil between them, plus a protective non-leaded glass plate on the cell side). Stations would be equipped with lights, utility distribution systems (electric and pneumatic), examination equipment, work tables, and up to two master/slave manipulators. The cell would be designed so that equipment could be added or removed from the work station without releasing radioactive contaminants, diluting the inert cell atmosphere, or extensively interrupting work at adjacent stations. The interior of the hot cell would be lighted, and high-intensity lighting would be provided in the cell at each active work station. In addition, emergency lighting would be provided.

Fuel material and test assembly storage would be available at various locations in and below each hot cell. There would be two 33-inch inner-diameter steel pits, extending below the level of the cell steel floor and the facility basement. These pits would be directly below the cell-roof loop-transfer penetrations for direct access. The pits can be covered when not in use.

The decontamination cell would be a shielded hot cell with an air atmosphere, maintained at a negative pressure relative to the surrounding corridors to minimize the spread of contamination. The decontamination cell would have six work stations and six leaded-glass observation windows. The decontamination cell would be separated from the inerted cell by an ordinary concrete shielding wall. The decontamination cell would be the same width and height as the inerted cell, and its outer walls would be similarly constructed. The cell floor would be lined with stainless steel, and the lower walls would be lined with carbon steel coated with epoxy paint. Electrical and pneumatic services in each decontamination cell would be generally similar to those in the inerted cell.

Support systems within the hot cell facility would be shared by the post-irradiation examination and spent fuel treatment processes.

The hot cell facility would have two distinct HVAC systems for contamination and emissions control: a cell exhaust system and a building/laboratory exhaust system. Both the cell and the building/laboratory ventilation exhausts would be HEPA-filtered.

Utility distribution systems supporting the hot cell facility include normal electrical power supplied by the commercial grid; optional standby electrical power supplied by two diesel generators; instrument and vital compressed air; fire, potable, and service water systems; and communications. Compressed gas for process applications would be supplied by standard compressed gas cylinders. Compressed argon for cell inerting would be supplied by a liquid argon tank system located outside the hot cell facility.

The control room for hot cell facility operations would be located on the operating floor. Local instrument alarm panels would be installed on, or in the vicinity of, the applicable equipment (e.g., hot cell work-station equipment, hot cell atmosphere-cooling and purification equipment, ventilation systems).

Existing Facilities

In addition to this new hot cell facility, existing facilities at ORNL would be used for supplemental and/or advanced post-irradiation examination for materials that do not require an inert environment. Hot cells within the Irradiated Fuels Examination Laboratory, Building 3525, and the Irradiated Materials Examination and Testing facility, Building 3025E, would be used to supplement the capabilities of the new post-irradiation examination facility. In addition, the Low Activation Materials Design and Analysis Laboratory (LAMDA) would be used for testing of low dose samples, samples that do not require hot cells for article examination. No modifications to the existing facilities would be required in support of the VTR post-irradiation examination of test specimens.

The Irradiated Fuels Examination Laboratory in Building 3525 is a Category 2[13] nuclear facility and contains six hot cells (including a scanning electron microscope cell, irradiated microsphere gamma analyzer cell, and a core conduction cool-down test facility cell) that are currently used for examination of a wide variety of fuels. The facility has been used for safety testing of High Temperature Gas Reactor fuel. Examination and testing capabilities include destructive and non-destructive testing of irradiated samples by techniques including metrology, optical and electron microscopy, gamma spectrometry, and other physical and mechanical property evaluation techniques (ORNL 2015).

The Irradiated Materials Examination and Testing facility in Building 3025E is a Category 3 nuclear facility that contains six hot cells (four of which are connected by transfer drawers) that are used for mechanical testing and examination of highly irradiated structural alloys and ceramics. The facility also includes a

[13] DOE defines hazard categories by the potential impacts identified by hazard analysis and has identified radiological limits (quantities of material present in a facility) corresponding to the hazard categories: Hazard Category 3 – Hazard Analysis shows the potential for only significant localized consequences; Hazard Category 2 – Hazard Analysis shows the potential for significant onsite consequences beyond localized consequences (DOE 2018a).

Specimen Prep Lab equipped with laboratory hoods and glove boxes. It is a two-story block and brick building with a two-story high bay (ORNL 2014).

LAMDA is a laboratory for the examination of materials with low radiological content (samples limited to less than 100 millirad per hour at 30 centimeters) that do not require remote manipulation. LAMDA capabilities focus on mechanical, physical, and microstructural characterization of samples. The LAMDA facility augments the capabilities in the ORNL hot cell facilities by adding a more precise and delicate sample-handling capability allowing for the study of material phenomenon not possible in a hot cell facility (ORNL 2017).

B.3.4.2 Environmental Resources – Construction

In addition to the resource requirements for the post-irradiation examination capability, these resources include the resources required for construction of the spent fuel treatment capability. Both capabilities are located within the same new facility; the Post-Irradiation Examination and Spent Fuel Treatment Facility. Estimates of environmental resources were developed for the facility, not each individual capability.

Resource Requirements

Table B–21 provides a summary of the key resources committed to the construction of the post-irradiation examination and spent fuel treatment capability. The construction effort would ramp up until peaking in the third year of construction. The resources required for site preparation have been included in the resource requirements for VTR construction at ORNL (see Section B.2.11).

Table B–21. Resource Requirements during Oak Ridge National Laboratory Post-Irradiation Examination and Spent Fuel Treatment Facility Construction

Resource	*Units*	*Annual Average Value*	*Annual Peak Value*	*Total* [a]
Staff	FTE	200	390	960
Electricity	kWh	300,000	600,000	1,300,000
Gasoline	gallons	26,000	44,000	110,000
Diesel Fuel				
Road Diesel	gallons	25,000	43,000	110,000
Non-road Diesel	gallons	130,000	230,000	570,000
Total Diesel	gallons	160,000	270,000	690,000
Water				
Potable	gallons	2,400,000	3,600,000	12,000,000
Dust Control, etc.	gallons	6,600,000	12,000,000	27,000,000
Total	gallons	9,000,000	16,000,000	39,000,000
Asphalt	cubic yards	420	NA	420
Structural Concrete	cubic yards	--	--	12,000
Rebar	tons	--	--	1,300
Excavation	bank cubic yards [b]	--	--	41,000
Backfill Material	cubic yards	--	--	60,000 [c]
Landscaping	cubic yards	--	--	600
Structural Steel	tons	--	--	1,200
Large Bore Piping	linear feet	--	--	9,500
Cable and Wire	linear feet	--	--	360,000
Cable Tray	linear feet	--	--	5,400
Conduit Above Grade	linear feet	--	--	66,000
Conduit Inside Duct Banks	linear feet	--	--	16,000
Rock/Gravel	cubic yards	--	--	14,000
Temporary Concrete	cubic yards	--	--	4,200
Lumber	tons	--	--	75
Temporary Steel	tons	--	--	15
Gas [d]	bottles/cubic meters	--	--	6,000/39,000

Resource	Units	Annual Average Value	Annual Peak Value	Total [a]

FTE = full-time equivalent (person); kWh = kilowatt-hour; NA = not applicable.

[a] Construction duration of 51 months is assumed.

[b] A bank yard is the volume of earth or rock in its natural state, as compared to the expanded volume after excavation.

[c] Excavated material would be temporarily stored within the construction footprint and would be used as backfill. Material from a borrow site would be used for the additional 19,000 cubic yards needed.

[d] Gas bottles (cylinders) can range from 2 to 10 cubic meters in size. A typical size of 6.5 cubic meters has been used to estimate the volume of gas in the cylinders.

Source: INL 2020c; Leidos 2020.

Nonradiological Releases

Nonradiological releases are associated with the operation of trucks and construction equipment (i.e., the burning of diesel fuel). Types and duration of operation for the equipment used during construction are discussed in the main body of this EIS. Emissions associated with equipment have been included in the estimates for construction of the VTR at ORNL in Table B–16.

Waste Generation

Table B–22 provides estimates of the wastes generated during facility construction. There would not be any radiological waste generated during construction of the Post-Irradiation and Spent Fuel Treatment Facility.

Table B–22. Oak Ridge National Laboratory Post-Irradiation and Spent Fuel Treatment Facility Construction Wastes

Waste Type	Material	Units	Value
Hazardous Waste			Assumed to be 2 percent of nonhazardous waste volumes
Nonhazardous Waste	Concrete	cubic yards	3,000
	Rebar	pounds	110,000
	Structural steel	tons	99
	Large bore pipe	feet	750
	Small bore pipe	feet	840
	Cable and wire	feet	29,000
	Cable tray	feet	420
	Conduit	feet	7,800
	Tubing	feet	840
	Instruments	each	20
	Valves	each	9
	In-line components	each	20
	Lumber	tons	36
	Steel	tons	15
	Gas	bottles	5,800

Source: INL 2020c; Leidos 2020.

B.3.4.3 Environmental Resources – Operations

In addition to the resource requirements for the post-irradiation examination capability, these resources include the resources required for operation of the spent fuel treatment capability. Both capabilities are located within the same new facility, the Post-Irradiation Examination and Spent Fuel Treatment Facility. Estimates of environmental resources were developed for the facility, not each individual capability.

The nominal test cycle length for the VTR would be 100 effective full-power days. At the end of each cycle, test assemblies at the end of their test exposure time would be removed from the core. The test

specimens within the assemblies would be allowed to cool within the reactor vessel for a period of time. When removed from the reactor vessel and after being cleaned (sodium removal), these test assemblies would be transferred to the post-irradiation examination facility.

Resource Requirements

Key annual resource commitments for the operation of the Post-Irradiation Examination and Spent Fuel Treatment Facility are provided in **Table B–23**. Diesel fuel would be required for testing of the site diesel generators.

Table B–23. Oak Ridge National Laboratory Post-Irradiation Examination and Fuel Treatment Facility Operational Resource Requirements

Resource	*Units*	*Value*
		Annual (peak)
Staff	FTE	100
Electricity	MWh	57,000 (60,000)
Diesel Fuel [a]	gallons	2,700
Water		
Potable	gallons	1,200,000
Component Wipedown	gallons	1,000
Total	gallons	1,200,000
Chemicals		
Acetone	pounds	15,000
Alcohol	pounds	30,000
Decon	pounds	14,000
Lubricant	pounds	1,400
Hydrochloric acid	pounds	1,000
Nitric Acid	pounds	17,000
Oil	pounds	2,300
Paint/Paint Thinner	pounds	1,800
Sodium Hydroxide Solutions	pounds	7,800
Gases		
Argon Liquid	standard cubic feet	61,000
Argon/Carbon Dioxide/Hydrogen/Methane/Methanol	liters	7,800
R-22 Refrigerant in Nitrogen/Air	liters	2,700

FTE = full-time equivalent (person); MWh = megawatt-hour.

[a] Diesel generators would operate 1 percent of the time, 88 hours per year.

Source: Leidos 2020.

Nonradiological Releases

Non-radiological releases result primarily from the testing of the building diesel generators and from the operation of personal vehicles by facility staff. The emissions associated with equipment have been included in the estimates for operation of the VTR at ORNL in Table B–17.

Radiological Releases

Radiological releases were estimated based on current releases from the HFEF and estimates of the gaseous inert fission products (INL 2020c) identified for examination of VTR test specimens. These estimates are presented in **Table B–24**. All releases from the facility would pass through HEPA filters (and from the main cell additional carbon filters) before being released through the facility stack.

Table B–24. Oak Ridge National Laboratory Post-Irradiation and Spent Fuel Treatment Facility Operational Annual Radiological Releases

Isotope	*A Post-Irradiation Examination Release (curies)*	*Spent Fuel Treatment Release (curies)*
Antimony-125	3.2×10^{-5}	1.57×10^{-7}
Americium-241	8.4×10^{-12}	
Carbon-14	3.1×10^{-4}	
Cadmium-109	5.2×10^{-4}	
Cadmium-113m		4.15×10^{-10}
Cadmium-115m	1.0×10^{-7}	
Cerium-144		1.41×10^{-6}
Chlorine-36	1.0×10^{-5}	
Cobalt-60	7.9×10^{-13}	2.08×10^{-9}
Cesium-134	8.0×10^{-7}	2.62×10^{-7}
Cesium-137	2.5×10^{-2}	1.96×10^{-6}
Europim-154		1.73×10^{-10}
Europium-155		2.07×10^{-9}
Iron-55		5.50×10^{-8}
Hydrogen-3 (Tritium)	3.7×10^{-2}	510
Iodine-129	1.8×10^{-5}	
Iodine-131	8.9×10^{-3}	
Krypton-85	4.4×10^{-3}	8.250
Neptunium-237	3.2×10^{-9}	
Niclel-63		2.76×10^{-10}
Promethium-147		1.25×10^{-7}
Phosphorus-32	2.6×10^{-5}	
Phosphorus-33	4.9×10^{-9}	
Plutonium-238	1.2×10^{-10}	1.24×10^{-10}
Plutonium-239	9.5×10^{-8}	2.83×10^{-9}
Plutonium-240	3.0×10^{-12}	1.87×10^{-10}
Plutonium-241		1.17×10^{-9}
Plutonium-242	1.8×10^{-9}	
Ruthinium-106		5.66×10^{-6}
Samarium-151		8.97×10^{-10}
Sodium-22	3.2×10^{-6}	
Sodium-24	1.7×10^{-8}	
Sulfur-35	1.2×10^{-4}	
Strontium-90	3.8×10^{-7}	3.47×10^{-8}

Source: INL 2020c.

Waste Generation

Annual waste generation rates for the Post-Irradiation Examination and Spent Fuel Treatment Facility are based on three VTR test cycles per year. These estimates are provided in **Table B–25**. This table includes waste generated from post-irradiation examination of test specimens, as well as spent driver fuel treatment. In addition to the wastes listed in this table, the heavy metal from 45 spent driver fuel assemblies (66 for the final core offload at the end of the VTRs operational lifetime) would be packaged as spent fuel.

Table B–25. Oak Ridge National Laboratory Post-Irradiation and Spent Fuel Treatment Facility Annual Operational Waste Generation

Waste Type	Category	Volume (cubic meters)		Weight (pounds)	
		Net	Gross	Net	Gross
Hazardous	NA	1.6	4.7	1,400	2,300
Industrial	NA	3.7	3.9	4,600	4,900
Recyclable	NA	1.2	1.2	1,900	2,000
TSCA	NA	0.053	0.054	70	87
Universal	NA	0.12	0.13	83	95
Low-level radioactive waste	Contact handled	220	240	110,000	160,000
	Remote handled	160	170	170,000	230,000
Mixed low level radioactive waste	Contact handled	16	21	21,000	25,000
	Remote handled	16	16	14,000	20,000
Transuranic waste	Contact handled	0.67	0.74	310	530
Mixed transuranic waste	Contact handled	0.14	0.15	62	100
	Remote handled	0.073	0.11	90	470

NA = not applicable.
Source: INL 2020d.

B.4 Spent Fuel Treatment and Storage

B.4.1 Introduction

Spent fuel would be stored within the VTR reactor vessel for about 1 year, until the decay heat produced drops sufficiently to allow for transport within a fuel transport cask and treatment of the spent fuel. Spent fuel treatment includes the removal of sodium from the spent fuel and the consolidation and packaging of the fuel. The fuel would be packaged in casks suitable for transport and storage at an onsite temporary storage facility and transport to and storage at a permanent repository.

Unless otherwise noted, information in the following subsections is from the *VTR Fuel Facility Plan* (INL 2019a).

B.4.2 Spent Fuel Treatment

The fuel would contain metallic sodium between the cladding and the metallic fuel pins to improve heat transfer from the fuel to the reactor coolant through the stainless-steel cladding. When fuel is irradiated in the reactor for some period of time, the metallic fuel swells as fission products are generated. Pores form throughout the fuel as it swells under irradiation and pressure from the gaseous fission products. The fission product gases escape to a plenum in the fuel element just above the metallic fuel. As the gases escape, liquid sodium flows into these tiny pores, much like a sponge. As more pores form and grow, others are closed off from the fuel surface, including those containing sodium. Between 20 and 40 percent of the available sodium (up to 0.8 grams) may enter the fuel and become inseparable from the uranium, except by dissolving or melting the fuel.

Maintaining a small inventory of untreated spent VTR fuel, perhaps 4 years or less of discharged fuel, would require that the fuel treatment facility treat fuel at the same rate as discharged by the VTR. These material throughput rates could be as high as 2.0 metric tons of fuel alloy per year with up to 1.8 metric tons of heavy metal per year.

The proposed treatment option for the sodium-bonded fuel elements would consist of five activities:

- Assembly disassembly,
- Fuel pin chopping,

- Consolidation and vacuum distillation of chopped fuel and plenums,
- Sodium stabilization, and
- Packaging.

Prior to transfer to the fuel storage pad, driver fuel assemblies would be washed at the VTR in a sodium wash station. At the wash station, the assembly would be washed inside of the wash station vessel by exposing the assembly to inert nitrogen gas containing demineralized water moisture. The demineralized water reacts with residual sodium to form sodium hydroxide. A second wash with demineralized water is used to remove the sodium hydroxide.

Up to six spent driver fuel assemblies would be transferred in a transfer cask to a spent fuel pad for temporary storage. The spent driver fuel assemblies would be inserted into a storage module within the interim dry storage system, where they would be stored for at least 3 years. (Three years would be the minimum storage time prior to spent fuel treatment and has been selected for planning purposes; the storage time could vary.) The interim dry storage system would consist of commercially available storage casks (INL 2020c).

Following the 3 additional years of cooling time, the spent driver fuel assemblies would be removed from the storage cask and transferred to a spent fuel treatment facility. All fuel treatment activities would take place in hot cells. VTR spent driver fuel assemblies would first be disassembled in the reverse of the assembly process described in Section B.5. Following disassembly, the fuel pins would be transferred to an element chopper.

Fuel pin chopping would consist of cutting the 165-centimeter fuel pins into much shorter pieces. Pieces free of spent fuel would be separated from pieces containing spent fuel. Gases released during the chopping process would be processed through a waste gas treatment system.

The container of chopped fuel would be placed into a vacuum distillation furnace. The entire driver fuel assembly (including reflectors and other smaller components) would be melted. Melting the full driver assembly would serve three functions: (1) reduce the concentration of the fissile material in the resulting consolidated product; (2) assist with fuel melting and consolidation; and (3) produce a more durable or corrosion-resistant, stabilized fuel product. The chopped segments of sodium-bonded fuel would be heated, evaporating the sodium, including the sodium that had migrated into pores in the fuel. The sodium-free fuel product (fuel, cladding, and possibly diluent) would continue to be heated to melt the product to form a eutectic[14] mixture, which would be removed from the furnace, solidified into ingots, and transferred to a packaging station. Individual ingots would weigh about 60 kilograms and would contain less than 10 percent by weight (no more than 6 kilograms) plutonium.

The sodium-free spent fuel ingots would be packaged in metal small canisters. The ingot canisters would have a robust metal shell and would fix the ingots into a location for criticality and transportation accident considerations. The ingot canisters would be filled with inert gas (argon or helium) and close-seal welded. A number of these canisters would be loaded into a DOE dual-purpose canister, providing an added measure of containment and protection for the spent fuel. The treated spent fuel would be loaded into a transfer cask, transferred back to the spent fuel pad, and transferred to the storage casks. Each storage cask would be capable of storing 120 ingots of treated spent fuel. This would be equal to 2 years of spent fuel generated by the VTR. The treated fuel would be stored onsite until an offsite storage capability (either a temporary storage site or a permanent repository) would be available (INL 2020c).

[14] A eutectic mixture is a homogenous mixture of two or more substances that solidifies at one temperature, lower than the temperature at which the individual substances solidify.

In the bottom section of the consolidation and distillation systems, sodium would be collected in a disposable steel container and transferred for stabilization. Depending on processing conditions, some volatile and semi-volatile fission products could be collected with the condensed sodium.

Fuel-pin plenum pieces (i.e., without fuel) would also be processed in a distillation system to remove any sodium but may or may not be consolidated into stainless-steel ingots. Sodium collected from the plenum sections would also be collected and transferred for stabilization.

Sodium stabilization would be achieved in a bakeout furnace. The sodium along with a stabilization chemical would be heated to about 800 degrees Celsius (°C) in a sealed steel shell. The stabilization chemical (possibly iron chloride) would react with the sodium to create a stable compound (e.g., combined with iron chloride, the reaction would produce iron and sodium chloride [salt]).

The sealed steel shells of stabilized salt and iron would be transferred to a packaging station where they would be placed in road-ready containers for shipment to a temporary waste storage location. Iron from sodium stabilization, sodium salt, and the processed plenums (sodium-free steel clads either as ingots or as scrap metal) would be treated as remote-handled low-level radioactive waste.

B.4.3 Spent Fuel Treatment and Storage at the Idaho National Laboratory Site

B.4.3.1 Idaho National Laboratory Facilities

All fuel treatment activities would be performed in the Fuel Conditioning Facility (FCF). The FCF is used to support nuclear energy research and development for multiple customers, including DOE, and is used to support the treatment of sodium-bonded spent fuel. (The FCF also supports developmental efforts in pyroprocessing; high-temperature chemical and electrochemical methods for the separation, purification, and recovery of fissile elements.) The FCF has two heavily shielded hot cells, one rectangular with an air atmosphere and one round with an inert (argon) atmosphere. Both are equipped with remotely operated manipulators to allow safe handling of irradiated fuels and materials. The inerted cell facilitates the preparation and treatment of spent fuel elements. Additionally, the facility has equipment to decontaminate and prepare elements for treatment, transfer components to other facilities (e.g., HFEF) and test, using mockup facilities, remotely operated systems designs (INL 2016).

To accommodate the material throughput identified in Section B.4.1, the FCF would require additional in-cell equipment treatment capacity, the replacement of a cell window to accommodate the transfer of spent driver fuel assemblies into the hot cell, and a transition to a 24-hour, 7-days-per-week operations schedule.

Fuel pin chopping would use existing FCF element choppers (see **Figure B–18**). In the existing element choppers, the linear slide feed mechanism is capable of handling up to five fuel elements of EBR-II fuel. Fuel pins are fed into the electromechanical press one at a time (INL 2020b). The press cuts them into elements that are between 0.25 and approximately 1.0 inches long (INL 2020b). For the VTR fuel pins, chopped fuel elements would be collected in separate baskets for fuel-containing elements and plenum elements. The FCF element choppers were designed to chop EBR-II fuel and have previously been modified to chop FFTF fuel and may need to be modified to accommodate VTR fuel pin length and diameter.

Spent fuel consolidation and distillation would use vacuum distillation furnaces. INL currently uses similar furnaces (see **Figure B–19**), in the HFEF. To handle the expected amount of spent fuel, multiple distillation units would need to be installed at the FCF. All fuel treatment actions would be performed in the argon atmosphere hot cell.

The sodium contaminated bakeout furnace would also be located within the FCF.

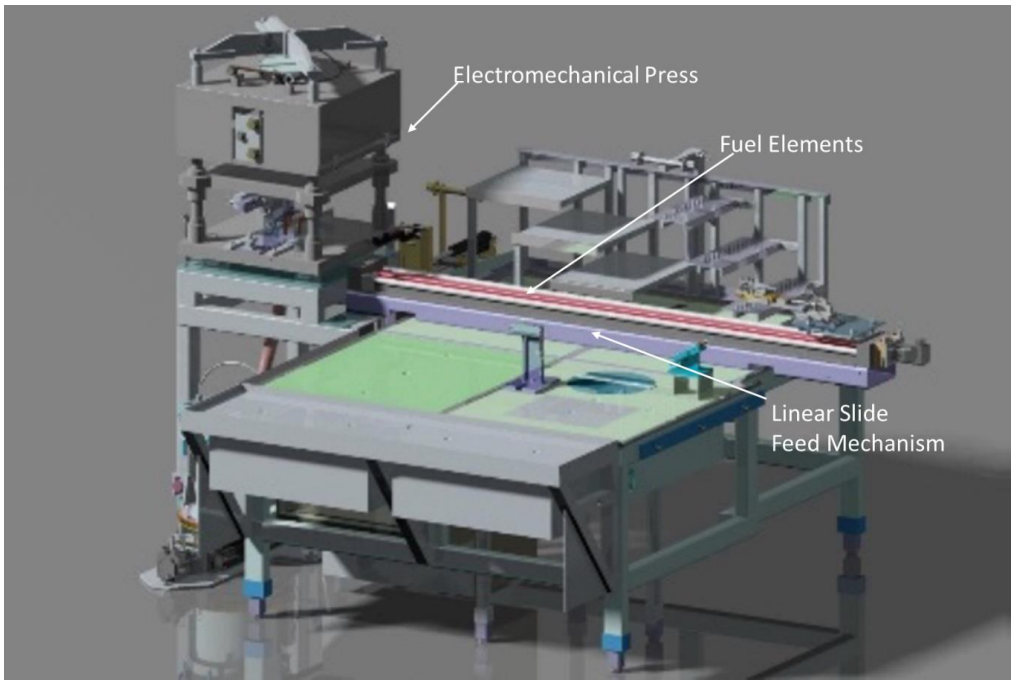


Figure B–18. Production Fuel Element Chopper in the Fuel Conditioning Facility

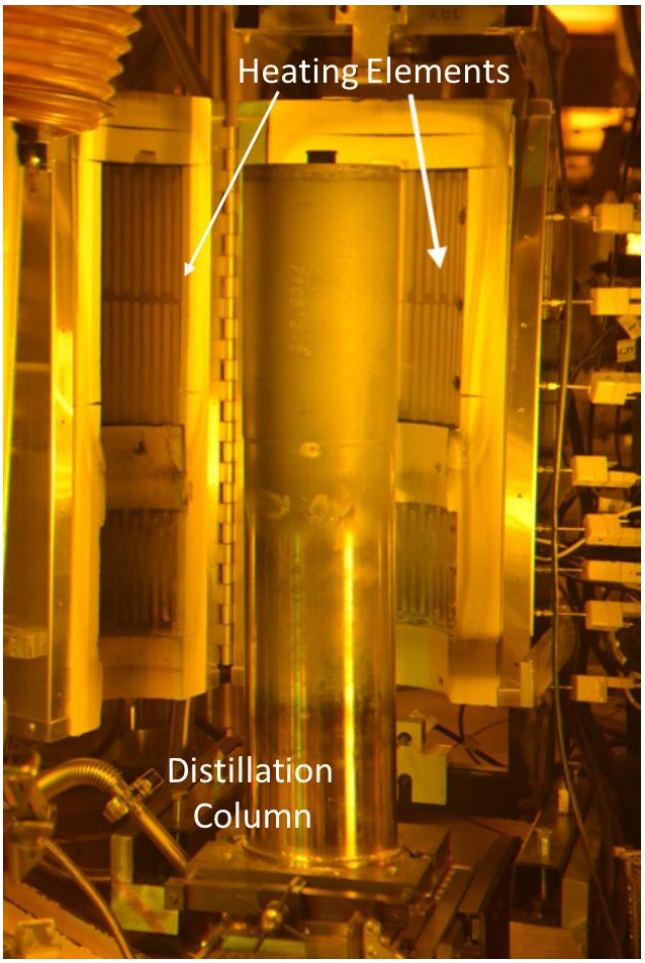


Figure B–19. Hot Fuel Examination Facility Distillation System

All products from the sodium treatment of the spent fuel would be packaged and temporarily stored (pending transfer to a permanent repository) at a facility at the MFC.

A new pad for the temporary storage of VTR spent driver fuel assemblies and treated spent fuel would be constructed on the VTR site at INL. The spent fuel pad could be required to handle all of the spent fuel from VTR operation (60 years) after treatment at the FCF.[15] Prior to the end of VTR operations, 3 years of spent fuel directly from the VTR would be stored on the pad. If sized to handle spent fuel from 60 years of VTR operations, the facility would consist of a concrete pad about 11,000 square feet (90 by 120 feet) and 4.5 feet thick. The spent fuel would be stored in qualified commercial storage casks (INL 2020c).

B.4.3.2 Environmental Resources – Construction

Resource Requirements

Resources required for the modifications to the FCF to accommodate VTR spent fuel treatment are limited to the workers needed to make the modifications and the use of potable water by these workers. INL estimates it would take a 10-person team working for 2 years to make the modifications. The workers would require 250,000 gallons of potable water during construction. Other material and utility use would be minimal.

Resource requirements for the construction of the spent fuel pad would be included in the construction of the VTR and its associated facilities. They would be a small fraction of that needed for the construction of the VTR (INL 2020c) and would not appreciably increase the resource requirements for construction of the VTR and its associated facilities.

Nonradiological Releases

Nonradiological emissions during the construction of the spent fuel treatment facility are expected to be minimal. Emissions from the construction of the spent fuel pad would not materially increase the emissions associated with construction of the VTR facilities.

Waste Generation

Replacement of an FCF hot cell window may be required to accommodate VTR fuel transfer into the hot cell. Should this modification be necessary, removal of the existing hot cell window would be expected to generate low-level waste: about 5.4 cubic meters (12,000 pounds) gross, 5.2 cubic meters (10,000 pounds) net. Construction of the spent fuel pad would result in minimal waste generation. Small amounts of excess concrete and rebar would be generated, which would be a small fraction of the waste generated from the construction of the VTR.

B.4.3.3 Environmental Resources – Operations

The nominal test cycle duration for the VTR would be 100 effective full-power days. At the end of each cycle, up to 15 spent driver fuel assemblies could be removed from the core (INL 2020c). The spent driver fuel assemblies would be allowed to cool within the reactor vessel for a period of time, nominally a year. When removed from the reactor vessel and after being cleaned (sodium removal), these spent driver fuel assemblies would be transferred to the spent fuel pad. After an additional cooling period, at least 3 years, these assemblies would be transferred to the Spent Fuel Treatment Facility within the FCF for treatment and consolidation. The resulting spent fuel waste form would be returned to and stored at the spent fuel pad until transferred to an offsite storage facility.

[15] The spent fuel pad could be smaller. The VTR program intends to ship spent fuel offsite as soon as an offsite storage option, either an interim storage facility or a permanent repository, is available.

Resource Requirements

Key annual resource commitments for the operation of the Spent Fuel Treatment Facility are provided in **Table B–26**. Only chemicals used in quantities of over 1,000 pounds are shown in the table. Other chemicals and gases would be used in smaller quantities (INL 2020d).

Table B–26. Annual Resource Requirements for Versatile Test Reactor Spent Fuel Treatment at the Fuel Conditioning Facility

Resource	*Units*	*Usage*
Staff	FTE	18 [a]
Electricity	kWh	8,300,000
Potable Water	gallons	230,000
Chemicals		
Alcohol	pounds	21,000
Acetone	pounds	14,000
Decon	pounds	14,000
Sodium hydroxide solutions	pounds	7,800
Oil	pounds	2,300
Paint/Paint thinner	pounds	1,800
Gases		
R-22 refrigerant in nitrogen/air	liters	2,700

FTE = full-time equivalent (person); kWh = kilowatt-hour.

[a] New staff; in addition, 66 current workers would be shared with existing programs.

Source: INL 2020c.

Nonradiological releases

The FCF is an existing operational facility at the MFC. The addition of VTR spent fuel treatment activities is not expected to increase the amount of nonradiological emissions from this facility.

Radiological Releases

Radiological releases were estimated based on current releases from the FCF. These estimates are presented in **Table B–27**. All releases from the facility would pass through HEPA filters (and from the main cell additional carbon filters) before being released through the facility stack. The combined flow rate would be about 34,900 cubic feet per minute at ambient temperatures. The release would be through a 60-inch diameter stack at an elevation of about 200 feet.

Table B–27. Idaho National Laboratory Spent Fuel Treatment Facility Operational Annual Radiological Releases

Isotope	*Curies*	*Isotope*	*Curies*
Antimony-125	1.57×10^{-7}	Krypton-85	8,250
Cadmium-113m	4.15×10^{-10}	Nickel-63	2.76×10^{-10}
Cerium-144	1.41×10^{-6}	Promethium-147	1.25×10^{-7}
Cesium-134	2.62×10^{-7}	Plutonium-238	1.24×10^{-10}
Cesium-137	1.96×10^{-6}	Plutonium-239	2.83×10^{-9}
Cobalt-60	2.08×10^{-9}	Plutonium-240	1.87×10^{-10}
Europium-154	1.73×10^{-10}	Plutonium-241	1.17×10^{-9}
Europium-155	2.07×10^{-9}	Ruthenium-106	5.66×10^{-6}
Iron-55	5.50×10^{-8}	Samarium-151	8.97×10^{-10}
Hydrogen-3 (Tritium)	510	Strontium-90	3.47×10^{-8}

Note: Only isotopes with a release of 1×10^{-10} curies or greater are listed.

Source: INL 2020d.

Waste generation

Annual waste generation rates for spent fuel treatment are based on the treatment of 45 driver fuel assemblies per year, a total of approximately 1.8 metric tons of heavy metal. These estimates are provided in **Table B–28**.

Table B–28. Idaho National Laboratory Spent Fuel Treatment Facility Annual Operational Waste

Waste Type	*Category*	*Volume (cubic meters)*		*Weight (pounds)*	
		Net	*Gross*	*Net*	*Gross*
Industrial	NA	1.8	2.0	4,600	4,900
Low-level waste	Contact handled	130	140	74,000	110,000
	Remote handled	160	170	170,000	230,000
Mixed low-level waste	Contact handled	10	12	13,000	15,000
	Remote handled	16	16	14,000	20,000

NA = not applicable.
Source: INL 2020d.

In addition to the waste identified here, the treated and conditioned fuel from 45 spent driver fuel assemblies, (previously identified as waste from the VTR) would be generated by spent fuel treatment. This treated fuel would be stored at the site until an offsite storage option (either an interim storage facility or a permanent repository when either becomes available for VTR fuel), at which time it would be shipped off site.

B.4.4 Spent Fuel Treatment and Storage at Oak Ridge National Laboratory

B.4.4.1 Oak Ridge National Laboratory Facilities

The storage and treatment of spent fuel at ORNL would require the construction of new facilities; no existing facilities at the site are capable of handling these activities. Spent fuel treatment of the VTR driver fuel assemblies requires the use of hot cells with an inert atmosphere. ORNL has no such hot cells. A conceptual design[16] for this facility has been developed to meet the process requirements identified in Section B.4.2, using equipment similar to that identified under the INL VTR Alternative in Section B.4.3. The spent fuel treatment activities would occur within the same facility envisioned for post-irradiation examination of test specimens (see Section B.3.4). Both the fuel treatment and temporary storage facilities would be located within the same protected area as the VTR.

The spent fuel treatment portion of the hot cell facility would have its own set of inerted hot and decontamination cells. The spent fuel treatment hot cell would be a concrete-shielded, steel-lined enclosure with interior dimensions of 30 feet wide by 70 feet long by 25 feet high. It would be filled with argon gas that provides an inert, non-oxidizing atmosphere. The associated decontamination cell would be a concrete-shielded, steel-lined enclosure with interior dimensions of 30 feet wide by 20 feet long by 25 feet high. It would be filled with air. The interior surfaces would be lined with steel. A raised steel floor would extend over part of the cell. Sections of the raised floor could be removed for access to the subfloor area. Test samples and equipment would be moved using two 5-ton cranes and electromechanical manipulators. The space beneath the removable floor would be used for storage; it would also house gas ducts and filters, and serve as additional space (depth) for vertical handling of long items.

There would be penetrations in the cell walls, roof, and floor for windows, utility service, feedthroughs, in-cell handling equipment, gas ducting, transfer hatches, etc. Penetrations into each cell would be steel-

[16] The conceptual designs have been developed for NEPA purposes only. This conceptual design is not as detailed as, nor is it to be considered, the conceptual design that is a part of the DOE facility design process.

lined, welded to the cell liner, and surrounded by high-density shielding closures or inserts. Closures or inserts for the penetration liners would have double seals, with the space between them pressurized with an argon purge.

The fuel treatment hot cell would have 15 work stations, each about 10 feet wide, equipped with a shielding observation window (layers of leaded glass with thin layers of mineral oil between them, plus a protective non-leaded glass plate on the cell side). Stations would be equipped with lights, utility distribution systems (electric and pneumatic), examination equipment, work tables, and up to two master/slave manipulators. The cell would be designed so that equipment could be added or removed from the work station without releasing radioactive contaminants, diluting the inert cell atmosphere, or extensively interrupting work at adjacent stations. The interior of the hot cell would be lighted, and high-intensity lighting would be provided in the cell at each active work station. Emergency lighting would also be provided.

The fuel treatment decontamination cell would be a shielded hot cell with an air atmosphere, maintained at a negative pressure relative to the surrounding corridors to minimize the spread of contamination. The decontamination cell would have six work stations and six leaded-glass observation windows. The decontamination cell would be separated from the inerted cell by an ordinary concrete shielding wall. The decontamination cell would be the same width and height as the inerted cell, and its outer walls similarly constructed. The cell floor would be lined with stainless steel, and the lower walls would be lined with carbon steel coated with epoxy paint. Electrical and pneumatic services in each decontamination cell would be generally similar to those in the inerted cell.

The spent fuel temporary storage facility would be similar to that proposed for use under the INL alternative, a concrete pad (see Section B.4.3.1).

B.4.4.2 Environmental Resources – Construction

Resource Requirements

Spent fuel treatment would be collocated in the same building as the post-irradiation examination capability at ORNL, the new Post-Irradiation Examination and Fuel Treatment Facility. Environmental resources associated with the construction of the Spent Fuel Treatment Facility have been included in the resources identified for the facilities used for post-irradiation examination of test specimens at ORNL (see Section B.3.4.2).

In addition to the spent fuel treatment capability, a spent fuel pad would be constructed at the VTR site at ORNL. The environmental resource requirements associated with this construction activity are presented in **Table B–29**.

Nonradiological Releases

Nonradiological releases are associated with the operation of trucks and construction equipment (i.e., the burning of diesel fuel). Types and duration of operation for the equipment used during construction are discussed in the main body of this EIS. Emissions associated with equipment have been included in the estimates for construction of the VTR at ORNL in Table B–16.

Waste Generation

Small amounts of waste would be generated during construction of the spent fuel pad. Waste would consist of 2 cubic yards of concrete and 10 cubic yards of municipal waste. It has been assumed that about 2 percent of this waste would be hazardous waste (Leidos 2020).

Table B–29. Oak Ridge National Laboratory Spent Fuel Treatment and Storage Facilities Construction Resource Requirements

Resource	*Units*	*Total*
Staff	FTE	8
Electricity	kWh	1,800
Gasoline	gallons	580
Diesel Fuel		
Road Diesel	gallons	35,000
Non-road Diesel	gallons	5,200
Total Diesel	gallons	40,000
Water		
Potable	gallons	100,000
Dust Control, etc.	gallons	NA
Total	gallons	100,000
Structural Concrete	cubic yards	2,700
Rebar	tons	72
Excavation	bank cubic yards [a]	4,700
Asphalt	tons	1,900
Backfill (rock/gravel)	cubic yards	4,600
Cable	linear feet	6,500
Conduit	linear feet	6,500
Fencing	linear feet	10,000
Isolation Area Rip Rap	cubic yards	12,200

FTE = full-time equivalent (person); kWh = kilowatt-hour; NA = Not Applicable.

[a] A bank yard is the volume of earth or rock in its natural state, as compared to the expanded volume after excavation.

Source: Leidos 2020.

B.4.4.3 Environmental Resources – Operations

The nominal test cycle duration for the VTR would be 100 effective full-power days. At the end of each cycle, up to 15 spent fuel assemblies could be removed from the core (INL 2020c). The spent fuel assemblies would be allowed to cool within the reactor vessel for a period of time, nominally a year. When removed from the reactor vessel and after being cleaned (sodium removal), these spent fuel assemblies would be transferred to the spent fuel pad. After an additional cooling period, at least 3 years, these assemblies would be transferred to the new Post-Irradiation Examination and Fuel Treatment Facility for treatment and consolidation. The resulting spent fuel would be returned to and stored at the spent fuel pad until transferred to an offsite location (either an interim storage facility or a permanent repository when either becomes available for VTR fuel), at which time it would be shipped offsite.

Spent fuel treatment would be collocated in the same building as the post-irradiation examination capability at ORNL, the new Post-Irradiation Examination and Fuel Treatment Facility. Environmental resources associated with the operation of the Spent Fuel Treatment Facility have been included in the resources identified for the facilities used for post-irradiation examination of test specimens at ORNL (see Section B.3.4.3).

B.5 Reactor Fuel Production

B.5.1 Introduction

The design of the VTR driver fuel assemblies was discussed in Section B.2.3. The driver fuel assembly and fuel pin designs are based on the most recent fuel designs for the EBR-II and metal fuel demonstrated in the FFTF. The VTR core would contain 66 driver fuel assemblies. These hexagonal assemblies would be approximately 3.85 meters in length and 11.7 centimeters wide (flat surface to flat surface). Each driver fuel assembly would contain a bundle of 217 fuel pins, upper and lower shield blocks, a grid to which the lower end plugs of the fuels are fixed and a surrounding hexagonal duct with upper and lower adaptors. Each of the fuel pins would be 1.65 meters long with a diameter of 0.625 centimeters. Within the fuel pin, there would be fuel slugs with a total length of 80 centimeters. The fuel pins would also have an 80-centimeter plenum (for a plenum-to-fuel volume ratio of approximately 1) filled with argon (and possibly a mixture of tag[17] gas isotopes) near atmospheric pressure. Upper and lower end plugs, made of the same material as the cladding, would be seal-welded to the cladding tube and the completed fuel pin would be helically wrapped with a spacer wire on a 15.2-centimeter (6-inch) pitch.

Ingot – an oblong metallic block consisting of one of the fuel elements; plutonium, uranium, and zirconium

Fuel slug – a cylindrical rod of alloyed fuel to be inserted into the fuel pin

Fuel pin – a single rod of fuel. The pin consists of a cladding tube, with top and bottom end plugs, containing fuel slugs, sodium-bonded to the cladding, and an inert gas plenum above the fuel.

Fuel assembly (sometimes referred to as a subassembly) – a hexagonal array of 217 fuel pins, top and bottom reflectors (shields) surrounded by an assembly duct with assorted mechanical components.

The metallic fuel (consisting of an alloy of uranium, plutonium, and zirconium) to be used in the VTR is unique and would be fabricated at a DOE facility separate from the VTR. Materials available for use in the production of the metallic fuel (feedstock) exist in several forms. Plutonium feedstock may be in the form of metals or oxides; uranium feedstock (of varying enrichments) may be in the form of metals, oxides, or nitrates. The fuel form for the fuel pin is a cast metallic cylindrical slug. The steps needed to convert these various feedstocks into VTR fuel would be:

- Conversion of feedstock from non-metallic forms to metals, if needed;
- Removal of impurities from feedstock, if needed;
- Fuel alloying and homogenization;
- Fuel slug casting and demolding;
- Assembly of the fuel slugs into fuel pins; and
- Assembly of the fuel pins into driver fuel assemblies.

The first two steps identified above would occur within a single facility, a feedstock preparation facility. The remaining steps would occur in a separate facility, the fuel fabrication facility. (If a single site were to be selected for both facilities, a single facility could be used to house both.) DOE has identified options for the siting of each of these activities, the INL Site and Savannah River Site (SRS). Separate sites could be selected for the two facilities; both could be located at the same site or either alone could be located at INL or SRS.

If sited at either INL or SRS, neither the feedstock preparation facility nor the fabrication facility would require the construction of a new facility, rather the equipment required would be installed within existing facilities (INL 2020c; SRNS 2020).

[17] Tag gas is a gas added to gas plenum used to help identify the location of any cladding leaks.

B.5.2 Versatile Test Reactor Fuel Production

The fuel needs for operation of the VTR were identified in Section B.2.3. Each year the VTR would need to replace up to 45 driver fuel assemblies. These assemblies would contain about 1,800 kilograms of fresh fuel; 400 kilograms of plutonium and 1,400 kilograms of uranium. Fuel production would require more than this amount of feed material to account for material left in the furnace during casting and rejected fuel rods (rods that do not meet fuel quality standards) that end up as fuel production waste. The efficiencies of the various fuel production operations vary, but as much as 27 percent of the fuel feedstock could end up as waste stream.[18] With this amount of feedstock becoming waste as much as 550 kilograms of plutonium and about 1,900 kilograms of uranium could be required to fabricate the 45 driver fuel assemblies per year. Over the 60-year lifetime of the VTR, this would result in the need for about 34 metric tons of plutonium and 120 metric tons of uranium feed material (SRNL 2020).

Not all of the plutonium available for the VTR exists in a form suitable for direct use in the driver fuel fabrication process. Preparation of the source material may be required to convert the plutonium into a metal and to remove impurities (polish) from the plutonium. Americium-241 is one of the primary elements targeted for removal, due to its impact on worker exposure.

Uranium is expected to be received in a form (metallic, acceptable impurity content) for use directly in the fuel fabrication process.

Feedstock Preparation

Feedstock preparation would address the first two steps in fuel production: conversion of feedstock from non-metallic forms to metals and fuel purification, removal of impurities. (Preparation is not anticipated to be required for uranium fuel feeds since metallic uranium fuel of the appropriate enrichment is commercially available.) There are several process options available for feedstock preparation. The selection of a preferred process methodology would depend upon, among other factors, the form and purity of the plutonium made available for the VTR program. Depending upon the form and quality of the plutonium feed, not all of the process steps described below may be necessary. It is even possible that plutonium with acceptably low impurity levels and in a metallic form could be available for the VTR. In that case feedstock preparation would not be necessary. In addition to the feedstock preparation processes described below, other preparation processes are available. Even within the processes described, potential variations could be utilized. A final determination of the processes that would be used for the VTR program has not been made.

Three potential feedstock preparation processes are under consideration for VTR feedstock preparation: an aqueous capability, a pyrochemical capability, and a combination of the two.[19] In the aqueous process, the plutonium feed (containing impurities) is dissolved in a nitric acid solution and put through a series of extraction and precipitation steps until a polished plutonium oxide is produced. The proposed process then converts the oxide to a metal in a direct oxide reduction process. (A potential variation of this process would be to precipitate the oxide with plutonium trifluoride and convert the cake to a mixture of plutonium dioxide and plutonium tetrafluoride that could be then reduced directly to plutonium metal, if adequate worker shielding could be provided.) In one form of the pyrochemical process (molten salt extraction [MSE]), the metallic plutonium feed is combined with a salt, the mixture is raised to the melting point, and an electrical current is passed through the solution. Impurities (such as americium) react with the salt and the purified plutonium is collected at the bottom of the reaction crucible. If the pyrochemical

[18] The highest percentage of feedstock material entering the waste stream would be associated with an option where no feedstock preparation would be necessary and no provisions were made to recapture some of the material that could otherwise end up in the waste stream. Other fuel production options could result in less waste and a smaller quantity of plutonium and uranium feedstock.

[19] Oher processing options are available, including; a trifluoride precipitation process and direct dissolution of plutonium/uranium alloys.

process were selected, a direct oxidation reduction process would also be required to convert plutonium dioxide feeds to plutonium metal. Either process (aqueous or pyrochemical) could be used to process unusable fuel from the driver fuel fabrication process. If a combination of the two processes were to be selected, a smaller aqueous line to prepare this reject fuel could be incorporated into the pyrochemical process.

Regardless of the feedstock preparation process, each step in the feedstock preparation process would take place within enclosures intended to protect workers and to help limit releases. At this stage in the design process, DOE envisions feedstock preparation being performed in gloveboxes.[20] The design for the feedstock preparation process is in an early stage of development, and hot cells may be a preferred alternative to gloveboxes to mitigate workforce exposure for some operations (SRNS 2020).

Aqueous Plutonium Processing

The aqueous process is the most mature of the three feedstock preparation processes being considered. It is also the process capable of handling the widest variety of feeds and the easiest to automate. Feed material for the aqueous process would consist of "new" feed material and scraps from the driver fuel fabrication process. Although not the only form of aqueous processing, the major steps, **Figure B–20**, in the aqueous process identified for use with VTR fuel production (SRNL 2020; INL 2020e) include the following:

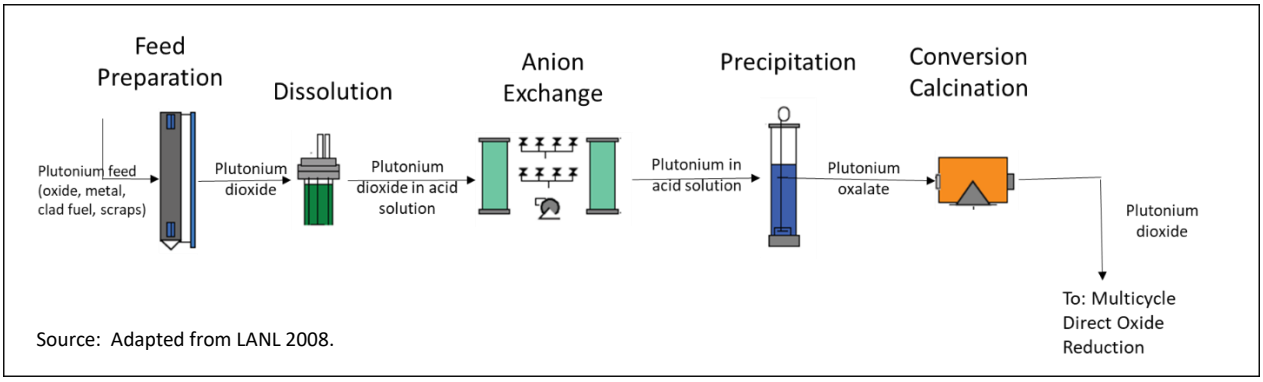


Figure B–20. Major Steps in Aqueous Processing

Feed preparation – Plutonium could be received in many forms: clad fuel or unclad material and in either an oxide or metallic form. The aqueous process works best with oxide feeds; dissolving metal feeds produces an unstable residue. Any feed material received in a metallic form would be converted to plutonium dioxide. Clad material would be processed to remove the cladding. The resulting materials would be ground to facilitate dissolution.

Dissolution – The plutonium dioxide would be dissolved in a strong nitric acid solution with other solvents (e.g. fluoride) and water. The resulting solution is filtered to remove any solid material (scrap).

Anion Exchange – The resulting solution is passed through an anion exchange column where a resin bed selectively absorbs the plutonium. The resin bed is an organic polymer that has positively charged sites imbedded in the solid polymer. Negatively charged mobile ions (in this case nitrates) balance the charge of the polymer. The resin preferentially captures the negatively charged plutonium in solution with the nitric acid, displacing the nitrates, while allowing impurities (americium, uranium, fluoride, etc.) to pass through the resin bed. The plutonium would be washed from the resin using a weak (nitric) acid solution.

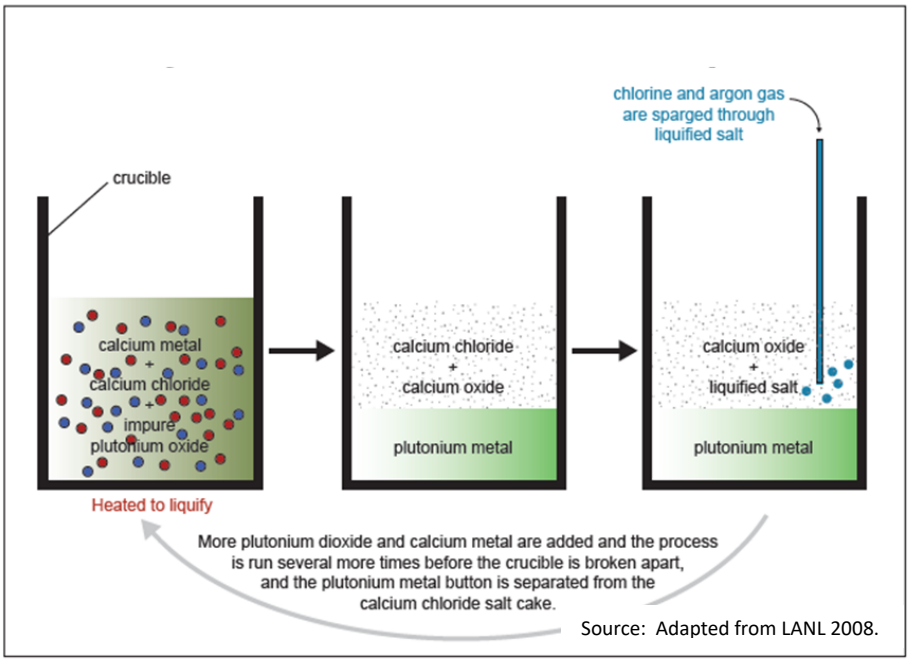
[20] Gloveboxes are sealed enclosures with gloves that allow an operator to manipulate materials and perform other tasks, while keeping the enclosed material contained. In some cases, remote manipulators may be installed in place of gloves. The gloves, glass, and siding material of the glovebox can be designed to provide worker radiation protection.

Precipitation – The product of the anion exchange is a weak acid solution that contains the purified plutonium. This solution is combined with another acid that reacts with the plutonium to produce an insoluble compound of plutonium, which is collected on a filter.

Conversion (*Calcination)* – The insoluble plutonium compound is put into a calciner, a vessel in which the plutonium is heated and dried. Oxygen is added to the calciner, reacting with the plutonium compound, creating plutonium oxide.

The plutonium oxide is the final product in the aqueous plutonium purification process. This product would be converted to metallic plutonium and cast into ingots for use in the fuel fabrication process.

Multicycle Direct Oxide Reduction (MDOR) – Direct oxide conversion (**Figure B–21**) converts oxide to metal feeds. The plutonium oxide is combined with a salt (calcium chloride) and calcium metal in a crucible within a furnace and heated to melt the mixture. The plutonium oxide and calcium react, producing plutonium metal and a mixture of calcium oxide and liquefied salt. As the mixture cools the plutonium metal (called plutonium buttons) collects at the bottom of the crucible. In a once-through process, the calcium/salt mixture retains a significant amount of the plutonium. However, the salt and calcium can be regenerated and reused in multiple oxide conversion cycles, thus reducing the amount of plutonium lost in the process.



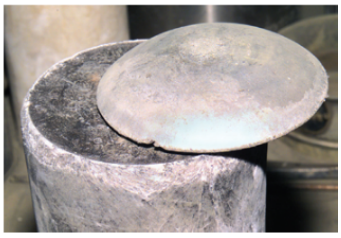

Plutonium Button

Figure B–21. Multicycle Direct Oxide Reduction

Casting – This final step in the feedstock preparation process produces the ingots for fuel fabrication. The output of the MDOR is vacuum cast into ingots in a furnace. The furnaces use a reusable crucible for melting, a coated graphite crucible to collect the casting, and are operated at 800 °C, under vacuum. This final step removes salt and slight impurities from the buttons.

Waste Handling – Radioactive waste is generated in most of the steps of aqueous and MDOR processing. Waste material from feed preparation and plutonium dissolution would have to be dried, oxidized, and downblended or immobilized (combined with an inert material). Liquid waste from anion exchange and precipitation would be processed to recover acids and the remaining waste would be solidified via evaporation. Each of these operations would require specialized equipment operated in gloveboxes. Crucibles from the MDOR and casting (collection of the plutonium products involves breaking the crucibles) would be wastes.

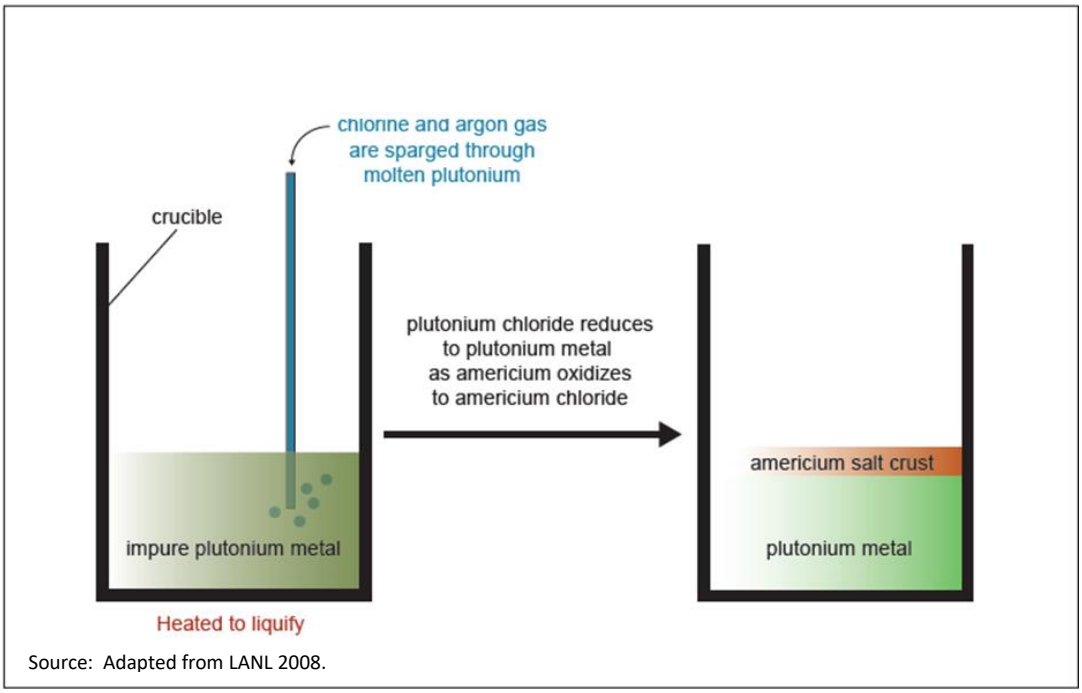
Pyrochemical and Electrorefining Plutonium Processing

The pyrochemical process would process the plutonium in metallic form rather than oxides needed for the aqueous process. The pyrochemical process has the advantage that fewer steps are involved in the purification process, and the entire operation would require less space than the aqueous process. However, the process identified for handling the majority of the plutonium does not handle feed material with higher impurity content as well as the aqueous process. Two separate processes would be utilized for VTR fuel. An MSE process would be used for "new" feed; an electrorefining process would be used for driver fuel fabrication product and "new" feed with higher impurity content.

The major steps in the preparation of the "new" plutonium feed by pyrochemical processing (SRNL 2020; INL 2020d) would include the following:

Feed Preparation – The same MDOR process described above would be used to convert any oxide feed to metal. Metallic feeds would not require any feed preparation.

Molten Salt Extraction (MSE) – In MSE (also called Metal Chlorination) (**Figure B–22**), plutonium metal is processed in batches with a salt. The mixture is heated to the melting point in a crucible, and chlorine gas is mixed with the molten mixture. This produces compounds of americium and plutonium, resulting in almost all of the americium and some of the plutonium being retained in the salt. The salt separates from the metallic plutonium, forming a salt crust that can be removed from the plutonium metal, and when mixing and heating is stopped the plutonium forms a button.



Source: Adapted from LANL 2008.

Figure B–22. Molten Salt Extraction

Vacuum Casting: Vacuum casting removes excess chloride and light metallic impurities, as described under the aqueous process. The resultant button is expected to be of sufficient purity to meet the VTR specification, without any further processing, provided the feeds were pure enough.

Waste Processing – MSE produces salt wastes (salts containing impurities such as americium and some plutonium) that would be processed with a salt scrub and salt oxidation and disposal – the scrub alloy process uses an aluminum-magnesium alloy to scrub the molten salt; impurities form a new alloy with the aluminum. The process removes most of the plutonium, essentially all of the americium, and produces a

scrub alloy ingot. Crucibles from the casting process would also be processed using the salt scrub. Waste salts would be oxidized and disposed as drummed waste.

Recyclable fuel fabrication products would be processed using an electrorefining process. In addition, "new" plutonium feed that contains a higher impurity content may need to be processed using electrorefining, due to the lesser ability of MSE to remove impurities. The major steps in the electrorefining process (SRNL 2020; INL 2020e) would include the following:

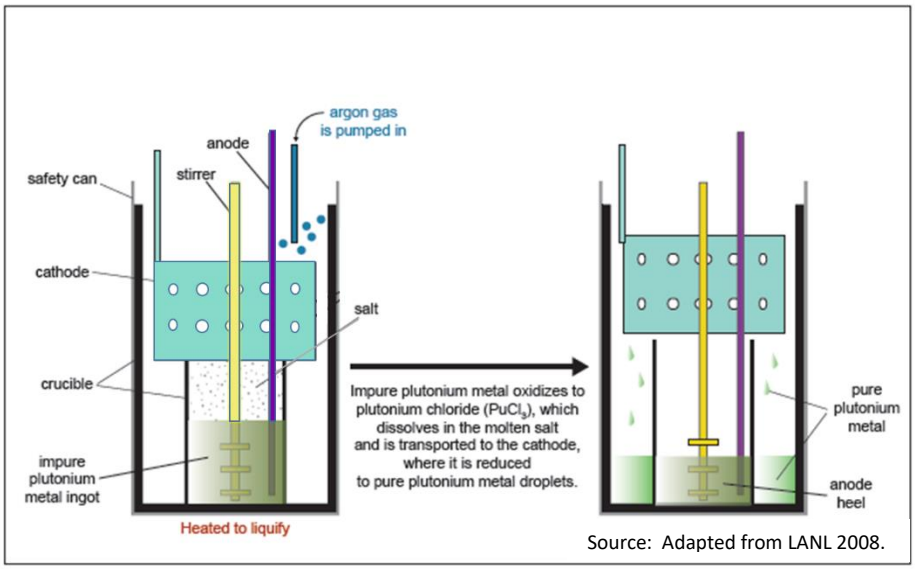
Feed Preparation – The material being dissolved would be chopped to increase its surface area. After chopping, the material would be placed in an anode basket and sent to the electrorefiner.

Electrolytic Reduction/Chlorination – "New" oxide feeds could be reduced to metal using either electrolytic reduction or chlorination. Electrorefining operates more efficiently when there are small quantities of metal chlorides in the salt mixture. A chlorination furnace could be included to produce these compounds as needed. Electrolytic reduction (essentially a single-step version of MDOR) could be used to prepare "new" oxide feeds for electrorefining. Electrolytic reduction could be used for oxide conversion, since the electrorefining process does not require the purity of feed material that the MSE process does.

Electrorefining – In the electrorefining process (**Figure B–23**), the chopped fuel is placed in a basket (or multiple baskets) in a molten salt. The basket acts as the anode (the negatively charged electrode) for the electrorefining process. A direct current is then passed between the anodes and cathodes (the positively charged electrodes), which dissolves and oxidizes the plutonium and uranium into the molten chloride salt. Multiple cathodes, at different electric potentials, allow deposition of uranium and plutonium metal onto different cathodes. In a typical arrangement, the anode is the inner section of a disc shape and the cathode is the outer ring of this disc shape.

Casting – Vacuum casting, similar to that used in the pyrochemical processing would be required to form the ingots used in fuel fabrication.

Waste Processing – Waste processing for the electrorefining process would be similar to that for the MSE process.



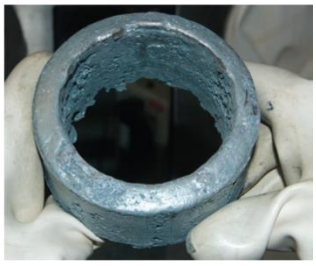

Metal plutonium ring

Figure B–23. Plutonium Electrorefining

Pyrochemical and Aqueous Plutonium Processing

The third feedstock preparation process utilizes the pyrochemical process (MDOR, MSE, and vacuum casting) in combination with a small aqueous line. The pyrochemical process would be used for "new" feeds, and the aqueous process would be used to further process products of the pyrochemical processing, as well as the unusable driver fuel. Additional processing would be required if impurity levels in the "new" feed plutonium are too high. A small electrorefining process line might be included in this option for these feeds. The processing steps would be the same as previously described for each process; although, the aqueous process would be on a smaller scale than needed if used to process all plutonium feeds.

Driver Fuel Fabrication

The driver fuel fabrication process takes the metallic uranium, plutonium, and zirconium metals and fabricates the finished driver fuel assemblies. Steps in the process include fuel alloying and homogenization, fuel slug casting and decasting, fuel pin assembly, and fuel assembly fabrication. Through pin assembly, these activities would occur in gloveboxes. (The design for the fuel fabrication process is in an early stage of development, and hotcells may be a preferred alternative to gloveboxes to mitigate workforce exposure for some operations (SRNS 2020)). Unless otherwise noted, information in this section is from the *VTR Fuel Facility Plan* (INL 2019a).

Fuel Alloying and Homogenization – An induction casting furnace would be used in the initial steps in the fuel fabrication process, alloying the elemental metallic components and producing the fuel slugs. (A possible design for the induction casting furnace is shown in **Figure B–24**.) This furnace would be contained within a glovebox with an inert gas atmosphere (see **Figure B–25**).

With the glovebox inerted, fuel constituents would be mixed together in their elemental metallic forms (i.e., as pre-weighed buttons, ingots, or chunks of uranium, plutonium, and zirconium) and melted together in a melt crucible to produce a chemically homogeneous uranium-plutonium-zirconium (U-Pu-Zr) alloy. This alloying and homogenization would take place in the casting furnace itself, without need for a separate fuel alloying process. The alloying step entails melting the alloy constituents and holding the melt at an alloying and homogenization temperature for some period of time. Inductive stirring in a U-10Zr melt has been shown to produce a homogenous mixture; however, for large batches of U-Pu-Zr, inductive stirring may not be sufficient to generate a homogenous mixture, and a tantalum stirrer may be required.

Fuel Slug Casting – The melted alloy of uranium, plutonium, and zirconium would be cast into cylindrical slugs by drawing the melt upward into quartz molds. The induction furnace glovebox would be evacuated to put all the molds under vacuum, and the furnace temperature would be adjusted from the homogenization temperature to the casting temperature. A mold palette (see **Figure B–26**), capable of producing about 135 fuel slugs,[21] would be preheated and then lowered into the melt crucible so every mold is dipped completely into the molten metal to a depth sufficient to keep the tips immersed in the melt throughout the casting process. The system would be rapidly pressurized to create a large differential pressure between the melt surface and the interior of the molds. The molds would then fill with molten metal. After sufficient time to allow the fuel alloy to begin to solidify within the molds, the mold palette would be raised to remove the molds from the melt.

[21] VTR operation would require the production of up to 19,530 usable fuel slugs per year when there are two fuel slugs per fuel pin. Initial plans call for two casting furnaces combined producing about four and a half batches per week (with 12 weeks of maintenance per year)resulting in the need (assuming non-recyclable and recyclable losses due to failed castings) for each batch to yield about 135 fuel slugs.

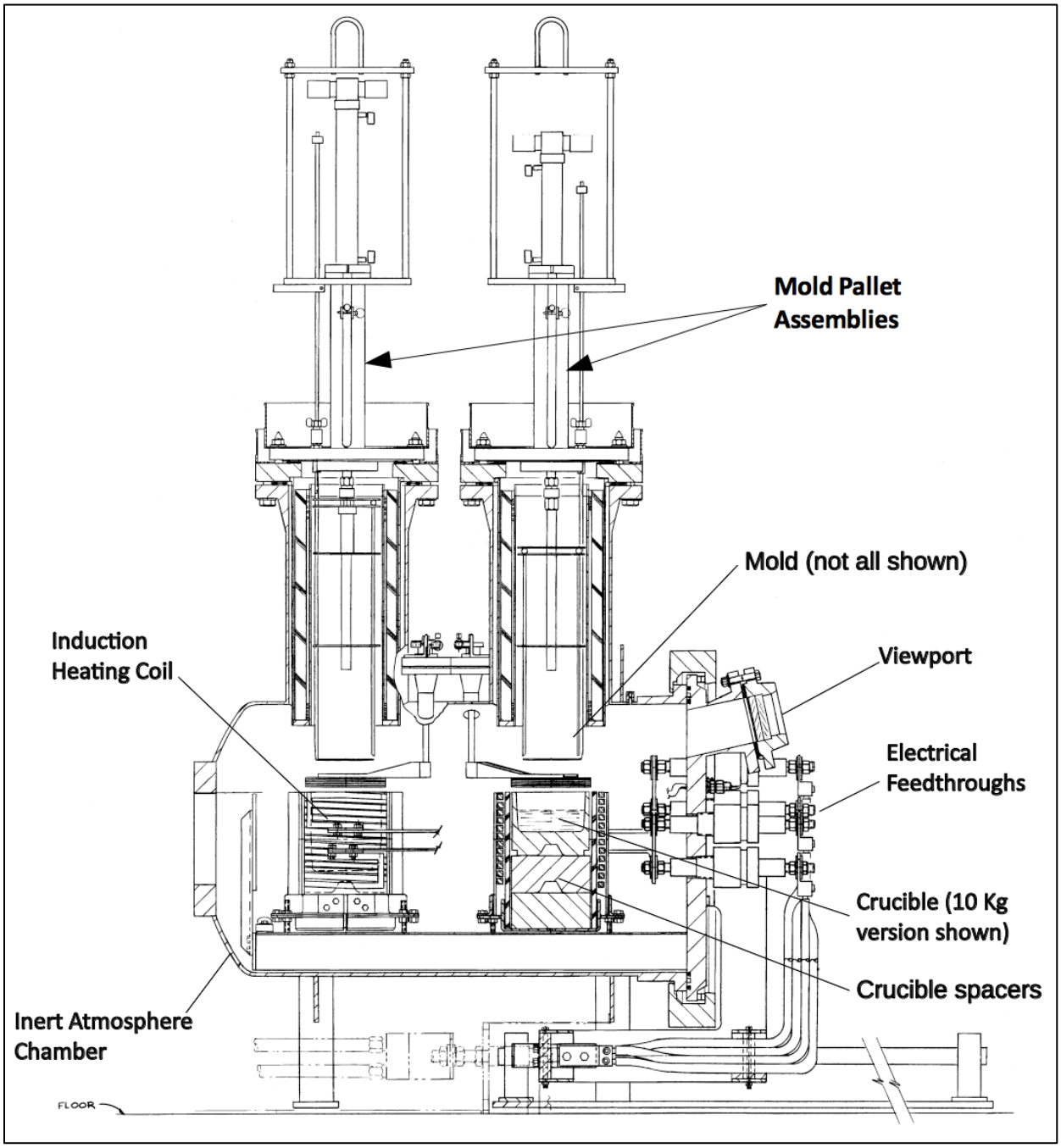


Figure B–24. Fuel Injection Casting Furnace

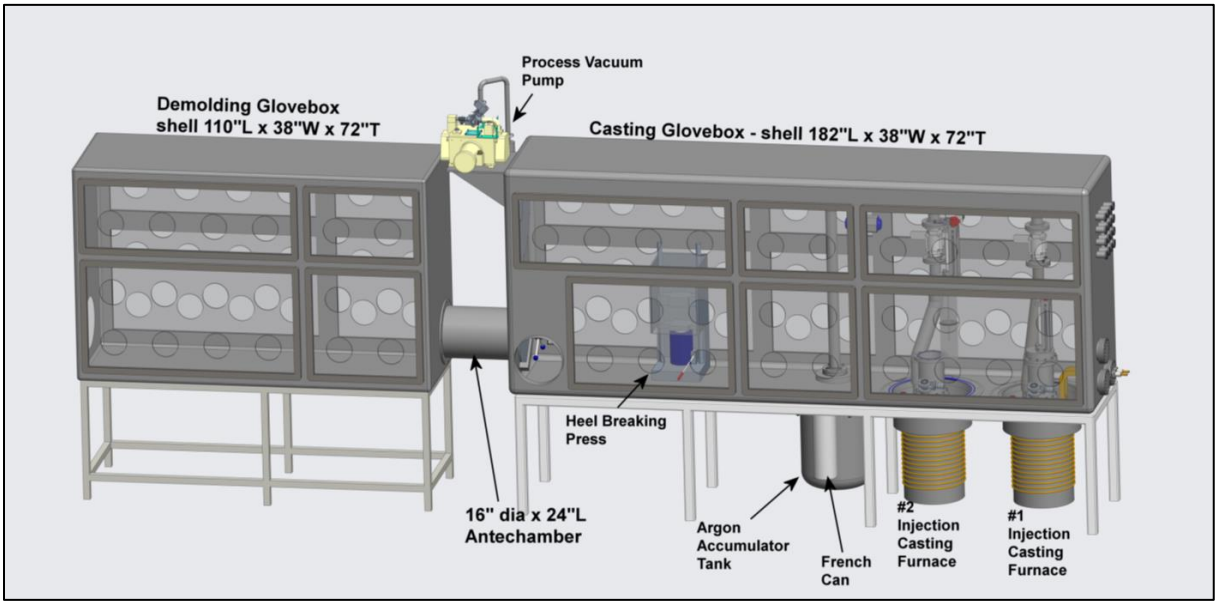


Figure B–25. Preconceptual Illustration of Slug Casting and Demolding Glovebox Line

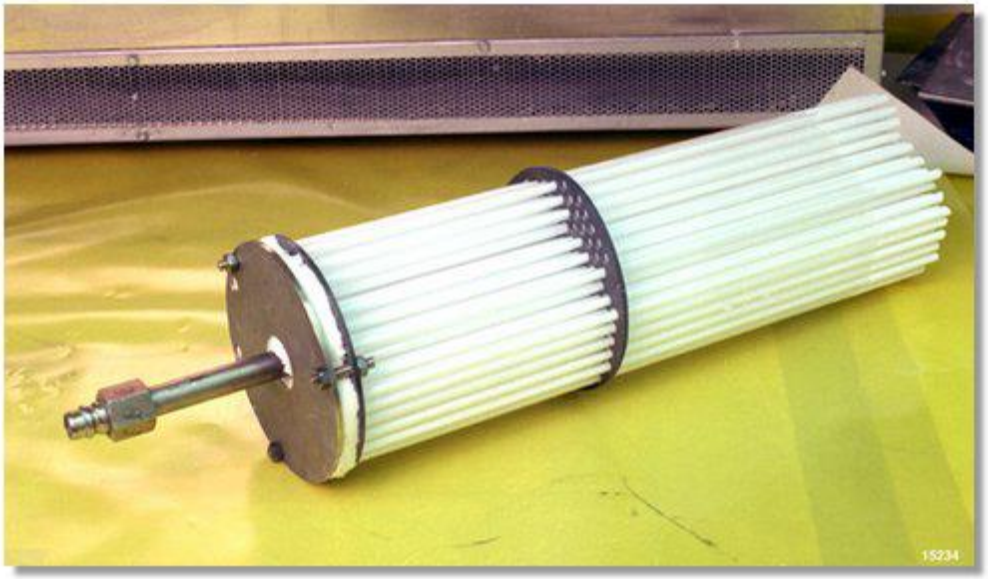


Figure B–26. Representative Casting Furnace Palette Ready for Loading into the Casting Furnace

Fuel Slug Demolding – In a separate glovebox with an inert atmosphere (see Figure B–25), the fuel slugs would be allowed to solidify and cool before removal. It may be possible to remove some fuel slugs through the palette hole, but it is expected that removing most slugs would require breaking the mold. Regardless of how the fuel slugs would be removed from the molds, molds are not reused. Once free of the mold, the fuel slug would be inspected for imperfections and surface defects. This function could be automated using machine vision to determine recoverable slug length, characterize any surface defects, and to determine straightness. Following inspection, the slug would be sheared to length, with final dimensions (length and diameter) measured by machine or manual inspection. Sheared material may be used for chemical analysis sampling, to determine that the fuel slugs meet specifications.

Fuel Pin Assembly – The prepared fuel slugs would be transferred to a third glovebox (see **Figure B–27**) with an inert atmosphere (argon with small amounts of helium) for fuel pin assembly. Fuel pin assembly

would consist of loading sodium (for bonding) and fuel slugs into a cladding jacket (a fuel pin cladding tube with the lower end plug welded into place).

After empty cladding jackets are introduced into the glovebox, extruded sodium metal would be inserted to slide to the bottom of the jacket. The amount of sodium inserted, when melted, would be sufficient to fill the fuel-cladding gap and provide a 2-centimeter cover above the top of the fuel. Fuel slugs, totaling 80 centimeters in length, would then to be inserted into the cladding jacket, to rest on top of the sodium. Finally, the top end plug would be pressed into the cladding jacket and the pin seal welded. The argon/helium glovebox atmosphere is the gas composition enclosed into the 80-centimeter fuel rod plenum. Helium would be included in the plenum gas to enable leak checking of the pin for a hermetic seal. After seal closure, fuel pins would be decontaminated and cleaned, which ensures that fuel pins can be handled outside the glovebox without plutonium contamination concerns. The final step in fuel pin assembly would be to wind the HT-9 steel, wire wrap spacer around the pin. The wrap would be welded to one end plug wrapped around the fuel pin and welded to the other end plug.

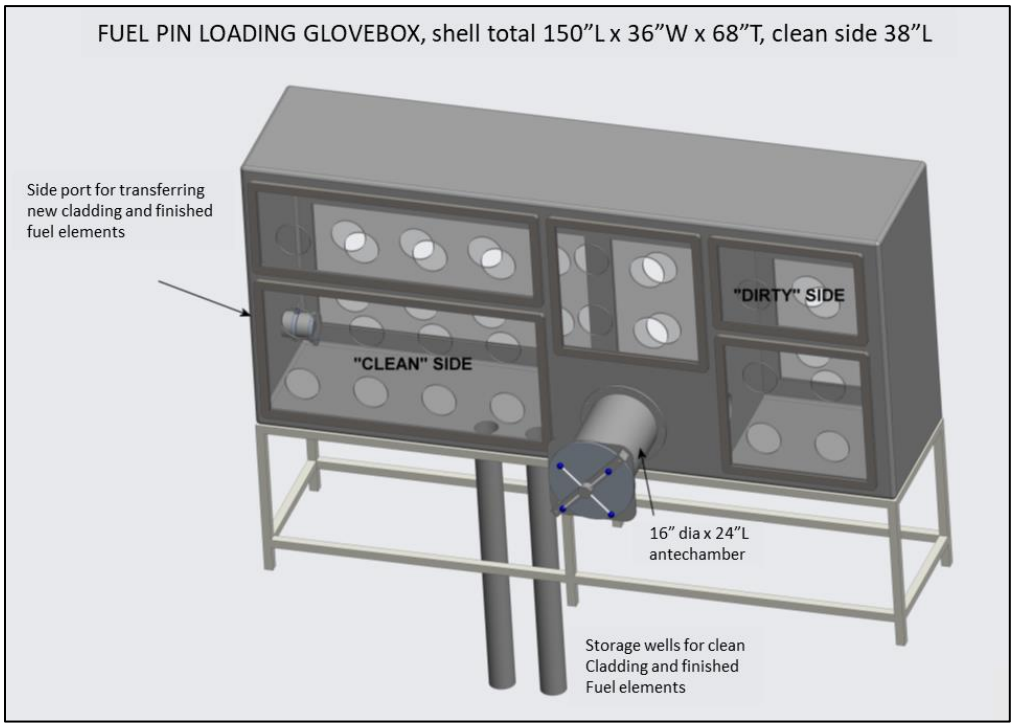


Figure B–27. Preconceptual Illustration of a Fuel Pin Loading Glovebox

Assembly Fabrication – The driver fuel assembly, described in Section B.2.3, would be fabricated using a Vertical Assembly Device, a fixture and loading station. The inlet assembly and lower shield block would be loaded into the device. A T-bar grid, providing proper spacing for the fuel pins, would be installed, and the fuel pins would be inserted into the grid such that the wire wraps properly intermesh. The upper shield block would be installed atop the fuel pins. Finally, the duct assembly (the duct, upper shield, and upper handling socket) would be inserted over the fuel pins and the duct would be secured to the assembly. The completed driver fuel assembly would be heated to melt the sodium filling the space between the fuel and cladding, providing a layer of sodium above the fuel slug. The assembly would be cooled and inspected, measured, and straightened, if needed. These operations would be carried out behind shielding; gloveboxes would not be required.

B.5.3 Idaho National Laboratory Site Reactor Fuel Production Options

Either or both feedstock preparation and driver fuel fabrication could be located at the INL site. Each option is described independently in the following sections. The equipment required for either process

could not be used for the other. However, there could be some benefit, in reduced resource use, in locating both options at the same site. In particular, construction resource use for both options may be less than the sum of resource use for the two options.

As described in the following paragraphs, DOE has identified existing MFC facilities that would be capable of supporting all fuel production activities. All of these facilities are currently in use and some (e.g., the ZPPR cell) have been identified as possible locations for future programmatic missions other than VTR reactor fuel production. Based on DOE programmatic and scheduling priorities, use of these facilities by other programs may result in their being unavailable to the VTR Program. Should this happen, modifications to enlarge an existing facility or the use of other MFC or VTR facilities would be evaluated to assess their capability to support the VTR Program. Any changes to the facilities being considered to host VTR reactor fuel production would be subject to review under the National Environmental Policy Act (NEPA).

B.5.3.1 Idaho National Laboratory Site Feedstock Preparation

B.5.3.1.1 Idaho National Laboratory Site Feedstock Preparation Overview

At the INL Site, this capability would be located in the FCF (a hazard category 2 facility[22]), but not in the FCF hot cells. Equipment would be installed in the Operating Floor/High Bay, the Mockup Area, and Workshop. Additionally, some space in the outer annulus of the FCF operating floor could possibly be repurposed for feedstock preparation. Equipment and operations currently located within this portion of the FCF would be relocated within the MFC. The identified area would be suitable for pretreatment operations like molten salt removal of the americium from plutonium (polishing) and direct oxide reduction and electrorefining to convert fuel compounds (e.g., fuel oxides) into their metallic form. The facility has space available that can be used to install the equipment required for these operations (INL 2020e).

At the current level of development for this process, designs for the glovebox have not been developed. Conceptually, they would be similar to gloveboxes currently used for plutonium processing. However, differences in size (based on processing rates) or the use of automation or other mechanisms to control worker dose would be expected.

Preparing the plutonium using the aqueous process (with direct reduction of the aqueous process plutonium dioxide product to plutonium metal) requires the largest area, and this process has been used to estimate the preparation area required. If the aqueous process is selected, the equipment required for feedstock preparation would consist of the following glovebox lines (INL 2020e):

- One line for feed preparation and product staging,
- Two lines for dissolving and adjustment,
- One line for anion reaction,
- Two lines for oxide conversion,
- One line for waste immobilization,
- Two lines for acid recycle and evaporators (2 lines approximately 60-foot each), and
- One line for accessory tanks.

[22] DOE defines hazard categories by the potential impacts identified by hazard analysis and has identified radiological limits (quantities of material present in a facility) corresponding to the Hazard Categories. Hazard Category 3: Hazard Analysis shows the potential for only significant localized consequences, Hazard Category 2: Hazard Analysis shows the potential for significant onsite consequences beyond localized consequences, DOE-STD-1027-2018.

Breakdowns for the arrangement of the gloveboxes for the pyrochemical process and for the combined pyrochemical/aqueous process have not been developed.

B.5.3.1.2 Environmental Resources – Construction

Construction activities associated with the feedstock preparation facility are limited to modifications to the FCF needed to convert space from its current purpose to feedstock preparation.

Resource Requirements

Resource commitments for the modification of the FCF to house the Feedstock Preparation Facility at INL are provided in **Table B–30**. In addition to the materials identified in this table, materials used in the construction of the gloveboxes include stainless steel for structural supports, glass for glovebox windows, piping for inlet, exhaust and other gas lines, electrical cable, and conduit for power and instrument lines. Primary gases used in the gloveboxes include argon as an atmosphere and hydrogen as a mechanism to remove oxygen from the glovebox atmosphere.

Table B–30. Idaho National Laboratory Feedstock Preparation Facility Construction Resource Requirements

Resource	*Units*	*Value*	
		Annual Average (peak)	*Total* [a]
For Modifications to Existing Facilities			
Staff	FTE	6 [a] (18 [b])	18
Electricity	kWh	Minimal [c]	Minimal
Diesel Fuel			
Forklift Fuel [d]	gallons	--	32
Mobile Crane Diesel [e]	gallons	--	120
Total Diesel	gallons	--	150
Water			
Potable	gallons	75,000	230,000
Construction Area Cleaning	gallons	1,700 (2,500)	5,000
Total	gallons	77,000	230,000
Propane, Butane		Minimal	Minimal
Gas (acetylene, oxygen)		Minimal	Minimal

FTE = full-time equivalent (person); kWh = kilowatt-hour.

[a] Construction duration of 3 years is assumed.

[b] Value represents peak number of workers at one time, not FTE.

[c] Electrical use is limited to hand held or cordless hand tools and occasional welding.

[d] Values assume 40 hours of operation and fuel consumption of 0.8 gallons per hour of operation.

[e] Values assume 30 hours of operation and fuel consumption of 4 gallons per hour of operation.

Source: INL 2020c.

Nonradiological Emissions

Nonradiological emissions during construction would be limited to emissions from personal vehicles and the cranes and forklifts used to move equipment. Emissions are presented in **Table B–31**.

Table B–31. Idaho National Laboratory Feedstock Preparation Facility Annual Nonradiological Releases During Construction

Table Calendar Year/Source Type	Air Pollutant Emissions (tons per year)						
	VOC	CO	NO_x	SO_2	PM_{10}	$PM_{2.5}$	CO_2e (metric tons)
Year 2024							
Onsite On-road Sources	0.000	0.02	0.002	0.000	0.000	0.000	3
Onsite Nonroad Sources	0.000	0.001	0.002	0.000	0.000	0.000	2
Offsite On-road Sources	0.001	0.13	0.01	0.000	0.004	0.001	16
Total Annual Emissions	**0.002**	**0.15**	**0.02**	**0.000**	**0.005**	**0.001**	**20**
Year 2025							
Onsite On-road Sources	0.000	0.02	0.001	0.000	0.000	0.000	3
Offsite On-road Sources	0.001	0.12	0.01	0.000	0.004	0.001	16
Total Annual Emissions	**0.001**	**0.13**	**0.01**	**0.000**	**0.004**	**0.001**	**18**

CO = carbon monoxide; CO_2e = carbon dioxide equivalent; NO_x = nitrogen oxides; $PM_{2.5}$ = particulate matter less than 2.5 microns in diameter; PM_{10} = particulate matter less than 10 microns in diameter; SO_2 = sulfur dioxide; VOC = volatile organic compound.
Source: Derived INL 2020d.

Waste Generation

Space within the FCF would be reallocated to support feedstock preparation. Equipment currently in this space would be relocated for use in other facilities. The removed equipment would not be waste. Waste generated during placement of the new equipment in the FCF would be minimal.

B.5.3.1.3 Environmental Resources – Operation

Resource Requirements

Key annual resource commitments for the operation of the feedstock preparation facility are provided in **Table B–32**. Resource requirements listed do not include the fuel feed material (uranium, plutonium, and zirconium).

Table B–32. Idaho National Laboratory Annual Feedstock Preparation Facility Resource Requirements

Resource	Units	Value	
		Annual	Peak
Staff	FTE	300	--
Electricity	MWh	6,700	--
Natural Gas	cubic feet	0	--
Heating Oil	gallons	0	--
Diesel [a]	gallons	1,500	--
Diesel (Operations)[a]	gallons	2,000	--
Water			
Potable Water [b]	gallons (thousands)	1,400	--
Process and Waste Treatment [c]	gallons (thousands)	50	--
Total	gallons (thousands)	1,500	--
Sanitary Waste Water Treatment	gallons (thousands)	1,400	--
Nitric Acid	cubic meters	88	130
Caustic	kilograms	43	64
Potassium Fluoride	kilograms	600	900
Aluminum Nitrate Nonahydrate	kilograms	300	450

Resource	Units	Value	
		Annual	Peak
Hydroxylamine Nitrate	kilograms	125	190
Polymer Resin	kilograms	40	60
Oxalic Acid	kilograms	1,400	2,100
Ascorbic Acid	kilograms	100	150
Argon	cubic meters	900,000	--
Helium	cubic meters	45,000	--
Nitrogen	cubic meters	50,000	--
Oxygen	cubic meters	5,000	--
Propane	bottles/gallons	100/470	150/700

FTE = full-time equivalent (person); MWh = megawatt-hour.

[a] Diesel fuel for one additional security vehicle and an additional diesel generator (Operations).

[b] Water use provided as gallons per minute, converted to annual assuming 8-hour work days, 5 days a week, and 50 weeks per year.

[c] Water requirements are for the aqueous processing of feedstock material. Other processes would require less.

Source: SRNS 2020.

Nonradiological Emissions

Nonradiological emissions for feedstock preparation would be associated with the transport of material to the FCF and worker vehicles. Emission data is presented in **Table B–33**.

Table B–33. Annual Nonradiological Operations Emissions from Feedstock Preparation Facilities at Idaho National Laboratory

Facility	Air Pollutant Emissions (tons per year)						
	VOC	CO	NO_x	SO_2	PM_{10}	$PM_{2.5}$	CO_2e (metric tons)
Haul Trucks	0.000	0.000	0.001	0.000	0.000	0.000	1
Worker Commuter Vehicles	0.003	0.39	0.03	0.000	0.01	0.002	48
Total Annual Emissions	**0.003**	**0.39**	**0.03**	**0.000**	**0.01**	**0.002**	**49**

CO = carbon monoxide; CO_2e = carbon dioxide equivalent; NO_x = nitrogen oxides; $PM_{2.5}$ = particulate matter less than 2.5 microns in diameter; PM_{10} = particulate matter less than 10 microns in diameter; SO_2 = sulfur dioxide; VOC = volatile organic compound.

Source: Derived from INL 2020c.

Radiological Releases

The VTR would require approximately 400 kilograms of plutonium each year, based on the need to replace 45 driver fuel assemblies per year. Depending upon the source of plutonium used as feed material for this process, the plutonium could contain varying quantities of impurities (especially americium-241). A representative estimate of the impurity content for the class of fuel containing the highest impurities was used to develop these estimate. Radiological releases were estimated assuming the feedstock preparation facility would process up to 580 kilograms of plutonium each year. This includes the processing of plutonium from driver fuel fabrication material (in a recycle of material unfit for use as VTR fuel) and plutonium that would be retained within wastes generated during feedstock preparation and fuel fabrication. The estimated annual release activity per isotope is presented in **Table B–34**.

Table B–34. Idaho National Laboratory Feedstock Preparation Facility Operational Annual Radiological Releases

Isotope	Release (curies)	Isotope	Release (curies)
Plutonium-238	9.5×10^{-6}	Uranium-232	5.8×10^{-12}
Plutonium-239	9.6×10^{-6}	Uranium-234	1.7×10^{-9}
Plutonium-240	1.4×10^{-5}	Uranium-235	1.5×10^{-11}
Plutonium-241	2.0×10^{-4}	Uranium-236	2.2×10^{-10}
Plutonium-242	2.2×10^{-8}	Uranium-238	4.39×10^{-11}
Americium-241	6.6×10^{-4}		

Note: Releases are based on processing 580 kilograms of plutonium and 460 kilograms of uranium each year.
Source: Adapted from SRNS 2020.

The HEPA-filtered releases of radioactivity to the environment would be through the existing FCF stack. The combined flow rate would be about 34,900 cubic feet per minute at ambient temperatures. The release would be through a 60-inch diameter stack at an elevation of about 200 feet.

Waste Generation

Annual waste generation rates, based on the steady-state production of about 45 driver fuel assemblies per year are provided in **Table B–35**. Estimated waste quantities for production (feedstock preparation and fuel fabrication) have been developed without considering any potential reduction in wastes that would result from the performance of both processes. In particular primary transuranic waste would not be doubled if both feedstock preparation and fuel fabrication were to be required. Estimated waste also may vary with the quality of the plutonium feedstock. The quantities listed here are expected to be representative of the waste generated during feedstock preparation.

Table B–35. Idaho National Laboratory Annual Feedstock Preparation Facility Operational Wastes

Waste Type	*Volume (cubic meters)*
Low-level radioactive waste	170
Mixed low-level radioactive waste [a]	2
Secondary transuranic	32
Mixed transuranic [a]	10
Primary transuranic	170
Hazardous – solid	1
Hazardous – liquid	1
Nonhazardous – solid	17
Nonhazardous – liquid	200
Universal	0.42

[a] For low-level and secondary transuranic radioactive wastes, the mixed waste volumes are included in the total waste.
Source: SRNS 2020.

B.5.3.2 Idaho National Laboratory Site Fuel Fabrication

The INL Fuel Fabrication Option includes the use of the FMF and the ZPPR to house the equipment necessary to support fuel alloying and homogenization, fuel slug casting, fuel pin assembly, and driver fuel assembly fabrication. VTR driver fuel fabrication is projected to require sample analysis for hundreds and potentially thousands of samples in the first few years of operation. INL proposes to use existing space fitted with new equipment in the FCF (Building MFC-765) as an analytical chemistry laboratory to support VTR fuel fabrication.

B.5.3.2.1 Fuel Fabrication Overview

Under this fuel fabrication option, the ingots of each fuel component (uranium, plutonium, and zirconium) would be delivered to the INL Fuel Fabrication Facility. At the INL Site, the Fuel Fabrication Facility would consist of existing INL facilities that would house the equipment needed to fabricate driver fuel assemblies from these ingots.

The driver fuel fabrication process at the INL Site would be located in the FMF and the ZPPR of the MFC (see Figure B–15). Both facilities are located within the MFC Protected Area, within its PIDAS. The FMF, adjacent to the ZPPR, consists of multiple workrooms and a material storage vault. The FMF has the ability to develop transuranic metallic and ceramic fuels, store these fuels, and produce and remove impurities from transuranic and enriched-uranium feedstock. The reactor and auxiliary systems portion of the ZPPR have been removed, and the facility is now used, among other tasks, for the storage, inspection, and repackaging of transuranic elements and enriched uranium. The ZPPR facility includes a workroom, cell area, and a material storage vault. As proposed, the three gloveboxes needed for fuel pin fabrication (casting furnace, demolding, and pin loading) and two additional gloveboxes for slug inspection and scrap recovery would be located in the south workroom of the FMF, where the existing Neptunium Repackaging-Transuranic Breakout Glovebox train is currently located. An existing uranium glovebox in this room would be removed. Two production lines are proposed (see **Figure B–28**). An existing glovebox train would be converted for use as one scrap recovery glovebox. The remaining casting gloveboxes, demolding gloveboxes, the train 2 scrap recovery glovebox, the slug quality assurance glovebox, and the pin loading glovebox shown in the figure would all be new equipment. Space in the MFC Special Nuclear Materials Storage Vaults would be prepared for material storage of:

- Plutonium feedstock,
- Fuel slugs,
- Fuel pins,
- Driver fuel assemblies, and
- Scrap and waste storage.

Space for lag storage of casting scrap, and assembled fuel pins pending transfer to ZPPR, would be made available in the FMF vault.

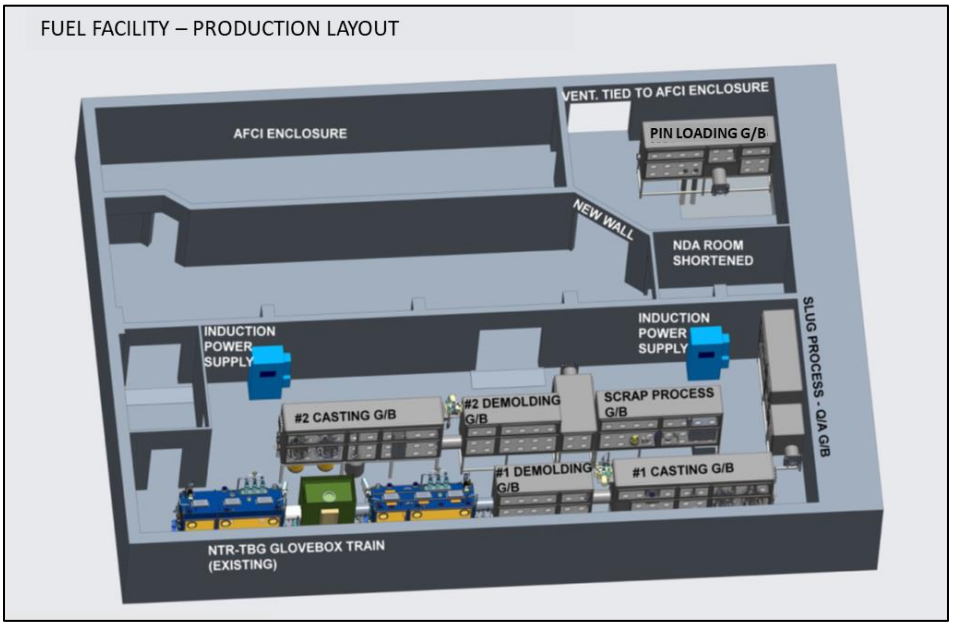


Figure B–28. Fuel Manufacturing Facility Fuel Pin Fabrication Equipment Arrangement

Upon completion of the fuel pin fabrication, fuel pins would be transferred to a storage vault or directly to the ZPPR reactor cell using a horizontal transport cask. Assembly of the fuel assembly, including bonding of the sodium to the fuel, would occur in the ZPPR reactor cell. New equipment would be installed to perform the following functions for assembly fabrication:

- Sodium bonding would be performed in a settling furnace,
- Fuel pins would be wrapped in an element (fuel pin) wire wrap station,
- Pin inspection would be performed using a profilometer and eddy current testing,
- Assembly fabrication would be performed in a vertical assembly device, and
- Assembly inspection would be performed in a vertical profilometer.

Additionally, temporary fuel pin storage racks, also located in the ZPPR reactor cell, would be required. Driver fuel assemblies could be stored in the ZPPR vault; this would require preparation of storage space, including installation of storage racks. The initial design objective for assembly storage would be sufficient capacity for 100 fresh assemblies, to ensure adequate supply for VTR operation, including the initial core load of 66 assemblies and most of the first year's reload fuel.

Driver fuel fabrication is projected to require sample analysis for hundreds and potentially thousands of samples in the first few years of operation. This workload, estimated as the analysis of 216 samples per week, and the required additional workspace would potentially overburden existing capabilities at the INL Analytical Laboratory (Buildings MFC-752). Additionally the plutonium content of samples would increase the radionuclide inventory of the Analytical Laboratory beyond the Hazard Category 3 limits currently in place. A revised safety analysis would be required to raise the facility to Hazard Category 2, before VTR fuel sampling could be done in the facility. This change would be potentially disruptive to current activities.

To minimize disruption to current activities, INL proposes to use existing space fitted with new equipment in the FCF (Building MFC-765) as an analytical chemistry laboratory to support fuel fabrication. Because the FCF is a Hazard Category-2 nuclear facility, the additional radionuclide inventory can be accommodated within the current hazard classification. **Table B–36** presents a list of equipment that would be needed to outfit the room.

Table B–36. List of Analytical Instrumentation Needed to Support Versatile Test Reactor Fuel Production

Equipment and instrumentation	*Purpose*
Class A TRU Glovebox	Manipulation of fuel samples (dissolution, dilution, disposition)
High Purity Germanium Detector System	Gamma spectrometry
Inductively Coupled Plasma-Optical Emission Spectrometer	Measurement of iron, cobalt, copper, nickel, beryllium, and other elements per fuel specifications
Ion Conductivity Probe	Measurement of chlorine and other accessible elements
Carbon, nitrogen, oxygen, and hydrogen analyzers	Light element analysis per fuel specifications
Multi-Collector – Inductively Coupled Plasma – Mass Spectrometer	High-precision measurements of uranium and plutonium isotopics (also possibly americium)
Nonradiological Fume Hood	Manipulation of nonradiological chemical reagents
Quadrupole Inductively Coupled Plasma – Mass Spectrometer [a]	Quantification of impurities per fuel specifications
Radiological Fume Hood	Preparation of dilutions and other manipulations

TRU = transuranic.

[a] Two instruments are recommended for high sample throughput and out-of-service contingency.

Initially, process qualification, development of a statistical understanding of the U-20Pu-10Zr as-cast fuel slug characteristics, and understanding phenomena such as elemental segregation during casting would

require a large number of samples. The number of analytical tests would decrease as the fuel fabrication process matured.

B.5.3.2.2 Environmental Resources – Construction

Metallic feed stock would be delivered to the FMF and no new facilities would be constructed at the INL Site. The only construction activities would be the build-out of the equipment locations in the FMF, ZPPR and FCF. Construction is assumed to require 2 years.

Resource Requirements

Table B–37 presents a summary of the key resources committed to the construction of a fuel fabrication facility. In addition to the materials identified in this table, materials used in the construction of the gloveboxes include stainless steel for structural supports, glass for glovebox windows, piping for inlet, exhaust and other gas lines, electrical cable, and conduit for power and instrument lines. Primary gases used in the gloveboxes include argon as an atmosphere and hydrogen as a mechanism to remove oxygen from the glovebox atmosphere.

Nonradiological Releases

Construction of the fuel fabrication facility and feedstock preparation facility would generate similar nonradiological emissions. The annual emissions associated with fuel fabrication facility construction would be the same as those presented in Table B–31.

Waste Generation

Wastes associated with fuel fabrication construction activities would be comprised of three main types: obsolete or replaced equipment, radiologically contaminated construction wastes, and cleaning supplies and clean wastes. These are anticipated to be minimal and consistent with current facility operations and existing NEPA documentation.

Table B–37. Idaho National Laboratory Fuel Fabrication Facility Construction Resource Requirements

Resource	*Units*	*Value*	
		Annual Average (peak)	*Total* [a]
For Modifications to Existing Facilities			
Staff	FTE	6 [a] (18 [b])	18
Electricity	kWh	Minimal [c]	Minimal
Diesel Fuel			
Forklift Fuel [d]	gallons	--	32
Mobile Crane Diesel [e]	gallons	--	120
Total Diesel	gallons	--	150
Water			
Potable	gallons	75,000	230,000
Construction Area Cleaning	gallons	1,700 (2,500)	5,000
Total	gallons	77,000	230,000
Propane, Butane		Minimal	Minimal
Gas (acetylene, oxygen)		Minimal	Minimal

FTE = full-time equivalent (person); kWh = kilowatt-hour.

[a] Construction duration of 3 years is assumed.

[b] Value represents peak number of workers at one time, not FTE.

[c] Electrical use is limited to hand held or cordless hand tools and occasional welding.

[d] Values assume 40 hours of operation and fuel consumption of 0.8 gallons per hour of operation.

[e] Values assume 30 hours of operation and fuel consumption of 4 gallons per hour of operation.

Source: INL 2020c.

B.5.3.2.3 Environmental Resources – Operations

The fuel fabrication facility would produce up to 19,530 usable fuel slugs per year when each fuel pin contains two fuel slugs, sufficient to supply up to 45 fresh driver fuel assemblies per year. A portion of the fuel slugs produced would not be expected to meet VTR fuel requirements. Most of the unusable fuel slugs could be processed in the feedstock preparation facility and would be recast into fuel slugs. However, some of the material would be expected to be captured in one of the fuel fabrication waste streams.

Resource Requirements

Key annual resource commitments for the operation of the fuel fabrication facility are provided in **Table B–38**. Only chemicals used in quantities of over 1,000 pounds are shown in the table. Other chemicals and gases would be used in smaller quantities (INL 2020d).

Table B–38. Idaho National Laboratory Fuel Fabrication Facility Annual Operational Resource Requirements

Resource	*Units*	*Value*
Staff	FTE	70
Electricity	MWh	8,300-13,300 [a]
Water		
Potable	gallons	880,000
Cleaning	gallons	1,000
Chemicals		
Alcohol	pounds	1,900
Nitric Acid	pounds	1,400
Gas		
Argon, compressed	standard cubic feet	30,000
Quartz	kilograms	3,000
Ytrria	kilograms	9
Zirconia Mold Wash	kilograms	90
Graphite	kilograms	500

FTE = full-time equivalent (person); kWh = kilowatt-hour.

[a] High and low values.

Source: INL 2020c; SRNS 2020.

Nonradiological Releases

Operation of the fuel fabrication facility and feedstock preparation facility would generate similar nonradiological emissions. The annual emissions associated with fuel fabrication facility operation would be the same as those presented in Table B–33.

Radiological Releases

Radiological releases were estimated assuming the fuel fabrication facility would process about 2,500 kilograms of uranium and plutonium. This quantity includes the material needed for the fuel product and some material that would be waste from fuel fabrication. The estimated annual release activity per isotope is presented in **Table B–39**. These releases assume the use of plutonium metal that either has been prepared as described in Section B.5.2.1, lowering any impurity content of the fuel to meet the VTR fuel quality criteria, or is from feedstock material that meets the VTR fuel quality criteria.

Table B–39. Idaho National Laboratory Fuel Fabrication Facility Operational Annual Radiological Releases

Isotope	*Release (curies)*	*Isotope*	*Release (curies)*
Americium-241	3.3×10^{-4}	Uranium-232	7.3×10^{-12}
Plutonium-238	2.3×10^{-6}	Uranium-234	2.2×10^{-9}
Plutonium-239	3.7×10^{-6}	Uranium-235	1.9×10^{-11}
Plutonium-240	2.4×10^{-6}	Uranium-236	2.8×10^{-10}
Plutonium-241	5.7×10^{-5}	Uranium-238	5.4×10^{-11}
Plutonium-242	1.7×10^{-9}		

Note: Releases are based on processing 550 kilograms of plutonium and 1,900 kilograms of uranium each year.
Source: Adapted from SRNS 2020.

The HEPA-filtered releases of radioactivity to the environment would be through the existing FMF stack. The combined flow rate would be about 6,400 cubic feet per minute at 64 °F. The release would be through a 36-inch diameter stack at an elevation of about 46 feet.

Waste Generation

Annual waste generation rates, based on the production of about 45 driver fuel assemblies per year are provided in **Table B–40**. The rates shown in the table are for the fabrication of fuel directly from feedstocks; feedstocks for which no feedstock preparation would be required. These feedstocks would contain impurities at levels below the acceptable limits for the VTR fuel. Should feedstock preparation be required, the transuranic wastes generated from fuel fabrication would be much less than the values shown in Table B-40. Other wastes would be generated in quantities similar to those shown.

Table B–40. Idaho National Laboratory Fuel Fabrication Facility Annual Operational Wastes

Waste	*Volume (cubic meters)*
Low-level radioactive	170
Mixed low-level radioactive [a]	2
Secondary transuranic	32
Secondary mixed transuranic [a]	10
Primary transuranic	170
Hazardous – solid	1
Hazardous – liquid	1
Nonhazardous – solid	17
Nonhazardous – liquid	200
Universal	0.42

[a] For low-level and secondary transuranic radioactive wastes, the mixed waste volumes are included in the total waste volume.
Source: SRNS 2020.

B.5.4 Savannah River Site Reactor Fuel Production Options

Either or both feedstock preparation and driver fuel fabrication could be located at SRS. Each option is described independently in the following sections. The equipment required for either process could not be used for the other. However, there could be some benefit, in reduced resource use, in locating both options at the same site. In particular, construction resource use for both options may be less than the sum of resource use for the two options.

Reactor fuel production capabilities could be installed in either the K Area Complex or the similar L Area Complex. The reactor buildings in K Area and L Area are of the same design, and like the K-Reactor Building, the nuclear fuel and equipment needed for reactor operations have been removed from the

L-Reactor Building. This EIS specifically evaluates the potential environmental impacts of using the K Area Complex in support of the VTR project, but the impacts would be similar if the L Area Complex were used. The reactor buildings are only 2.5 miles apart and each is within a PIDAS. At either location, activities would largely occur indoors with small, previously disturbed locations outside being used for construction laydown areas or for the construction of HVAC and entry control structures. At L Area, the option exists to use either the minus-20- and minus-40-foot levels or the ground floor level for reactor fuel production. A comparative analysis shows that the offsite impacts from radiological releases would be within 3 percent of each other, with those from L Area being slightly lower.

The description that follows assumes installation of reactor fuel production capabilities at K Area. A notional equipment configuration was developed to assess the capability to house the fuel production equipment within the identified structures. But, the equipment layout that would be used has not been determined and would be finalized during the detailed design of the fuel production facility.

B.5.4.1 Feedstock Preparation

B.5.4.1.1 Savannah River Site Feedstock Preparation Overview

At SRS, this capability would be located adjacent to the location for the driver fuel fabrication capability, in the K-Reactor Building (105-K) or the 108-K buildings in the K Area Complex, mostly at the minus-20-foot level (20 feet below grade).[23] About 10,000 square feet of space would be required for feedstock preparation in either location**,** The identified area would be suitable for pretreatment operations like molten salt removal of the americium from plutonium (polishing), electrorefining, and direct oxide reduction to convert fuel compounds (e.g., fuel oxides) into their metallic form.

As discussed for feedstock preparation at the INL Site, a design of the equipment for the feedstock preparation process has not been developed. A conceptual layout for the aqueous process, using the same glovebox lines as described for feedstock preparation at INL, would require the largest amount of space of the three processes being considered. (This is one possible layout other layouts are being considered.) This process fits within the available space at the K-Reactor Building, even if the fuel fabrication process is collocated within this structure.[24] To accommodate the feedstock preparation equipment, facility modifications would be required, including the addition of a new 8,000 square foot structure to house an upgraded HVAC system. This structure could be contained within one of the 108-K buildings, placed on top of one of the buildings or located adjacent to the structures on less than 3 acres of previously disturbed land within the K-105 Reactor Building security area, depending on whether one or both of feedstock preparation and fuel fabrication were to be located at SRS.

Most of the aqueous process equipment would be located at the minus-20-foot level; the plutonium dioxide to plutonium metal conversion equipment (the pyrochemical cell) would be located at the minus-40-foot level.

Breakdowns for the arrangement of the gloveboxes for the pyrochemical process and for the combined pyrochemical/aqueous process have not been developed.

B.5.4.1.2 Environmental Resources – Construction

Resource Requirements

Key annual resource commitments for the modifications in the K-Reactor Building to enable its use as the feedstock preparation facility are provided in **Table B–41**. In addition to the materials identified in this

[23] The location of the 108-K Buildings relative to the 105-5 Reactor Building is shown in figures provided in the discussion of fuel fabrication at SRS, Section B.5.4.2.

[24] The layouts for feedstock preparation and driver fuel fabrication depicted in this appendix were developed independently, neither considers the location of the other activity. The layouts would differ if both activities were to be located at SRS. However, there is sufficient space that both activities could be located within the K-Reactor Building structures.

table, materials used in the construction of the gloveboxes include stainless steel for structural supports, glass for glovebox windows, piping for inlet, exhaust and other gas lines, electrical cable, and conduit for power and instrument lines. Primary gases used in the gloveboxes include argon as an atmosphere and hydrogen as a mechanism to remove oxygen from the glovebox atmosphere.

Table B–41. Savannah River Site Feedstock Preparation Facility Construction Resource Requirements

Resource	*Units*	*Value*	
		Annual	*Total*
Staff	FTE	120	360
Electricity	MWh	minimal	minimal
Diesel	gallon	1,500	4,500
Gasoline	gallon	2,500	7,500
Water Supply			
Potable	gallons (thousands)	1,000	3,000
Construction	gallons (thousands)	2,000	6,000
Total	gallons (thousands)	3,000	9,000
Waste Water Treatment	gallons (thousands)	1,000	3,000
Cement	tons	--	800
Steel (tons)	tons	--	600
Conduit	linear feet	--	74,000
Cable Tray	linear feet	--	2,400
Power/Control Cable	linear feet	--	83,000
Piping	linear feet	--	14,000
Facilities	square feet	--	40,000
Ductwork	pounds	--	51,000
Formwork	square feet	--	36,000
Sand, Cone, Aggregate	cubic yards	--	880
Gravel, Crushed Stone, etc.	cubic yards	--	660
Soil – Fill Material	cubic yards	--	3,700
Gases			
Acetylene	cubic meters	--	53
Oxygen	cubic meters	--	240
CO_2/Argon	cubic meters	--	80
Nitrogen	cubic meters	--	160
Argon	cubic meters	--	1,300
Helium	cubic meters	--	33
Other			
Epoxy Floor Covering	square feet	--	48,000
Macropoxy (concrete wall covering)	square feet	--	17,000
Enamel Paint	square feet	--	50,000
Intumescent Coating (steel deck coating)	square feet	--	8,300

CO_2 = carbon dioxide; FTE = full-time equivalent (person); MWh = megawatt-hour.
Source: SRNS 2020.

Nonradiological Releases

Nonradiological releases are associated with the operation of the forklifts, construction vehicles, concrete mixers, cranes and other smaller equipment (i.e., the burning of diesel fuel and worker personal vehicle use). The total construction related emissions associated with these items are provided in **Table B–42**.

Table B–42. Savannah River Site Feedstock Preparation Facility Construction Nonradiological Emissions

Facility	Emissions (tons)							Combined HAPs [a]	CO_2e (metric tons)
	VOC	CO	NOx	SO2	PM_{10}	$PM_{2.5}$	CO_2		
Onsite Emissions from On-road Sources	0.02	1.62	0.20	0.002	0.05	0.01	221	0.005	201
Onsite Emissions from Nonroad Sources	0.04	0.85	0.24	0.001	0.02	0.01	63	0.01	57
Offsite Emissions from On-road Sources	0.05	3.24	0.44	0.003	0.10	0.02	458	0.01	416
Total 2025 Emissions	**0.11**	**5.71**	**0.88**	**0.01**	**0.17**	**0.05**	**742**	**0.02**	**674**

CO = carbon monoxide; CO_2= carbon dioxide; CO_2e = carbon dioxide equivalent; HAPs = hazardous air pollutants; NO_x = nitrogen oxides; $PM_{2.5}$ = particulate matter less than 2.5 microns in diameter; PM_{10} = particulate matter less than 10 microns in diameter; SO_2 = sulfur dioxide; VOC = volatile organic compound.

[a] Combined HAPs = 15/3 percent of combustive VOC/PM emissions for on-road and nonroad sources and 1/3 percent for slash burning (California Air Resources Board 2018).

Source: Adapted from SRNS 2020.

Waste Generation

Areas within the K-Reactor Building structures would be modified to make room for the feedstock preparation equipment. This would involve the removal of existing equipment and some structural modifications. Estimates for waste generation from this modification effort are shown in **Table B–43**.

Table B–43. Savannah River Site Feedstock Fabrication Facility Construction Wastes

Waste Type	Units	Value
Toxic Substance Control Act Waste	cubic meters	28
Universal Waste	cubic meters	7.5
Nonhazardous Waste		
From Construction Activities	gallons/cubic meters	90,000/340
Equipment Removed	metric tons/cubic meters	100/5,000
Low-level Radioactive Waste	cubic meters	380

Source: SRNS 2020.

B.5.4.1.3 Environmental Resources – Operations

Resource Requirements

Key annual resource commitments for the operation of the feedstock preparation facility are provided in **Table B–44**. Resource requirements listed do not include the fuel feed material (uranium, plutonium, and zirconium.)

Table B–44. Savannah River Site Annual Feedstock Preparation Facility Resource Requirements

Resource	Units	Value	
		Annual	Peak
Staff	FTE	300	--
Electricity	MWh	6,700	--
Natural Gas	cubic feet	0	--
Heating Oil	gallon	0	--

Resource	Units	Value	
		Annual	Peak
Diesel (Centerra) [a]	gallon	1,500	--
Diesel (Operations)[a]	gallon	2,000	--
Water			
Potable [b]	gallons (thousands)	1,400	--
Process and Waste Treatment [c]	gallons (thousands)	50	--
Total	Gallons (thousands)	1,500	--
Sanitary Waste Water Treatment	gallons (thousands)	1,400	--
Nitric Acid	cubic meters	88	130
Caustic	kilograms	43	64
Potassium Flouride	kilograms	600	900
Aluminum Nitrate Nonahydrate	kilograms	300	450
Hydroxylamine Nitrate	kilograms	125	190
Polymer Resin	kilograms	40	60
Oxalic Acid	kilograms	1,400	2,100
Ascorbic Acid	kilograms	100	150
Argon	cubic meters	900,000	--
Helium	cubic meters	45,000	--
Nitrogen	cubic meters	50,000	--
Oxygen	cubic meters	5,000	--
Propane	bottles/gallons	100/470	150/700

FTE = full-time equivalent (person); MWh = megawatt-hour.

[a] Diesel fuel for one additional security vehicle (Centerra) and an additional diesel generator (Operations).

[b] Water use provided as gallons per minute, converted to annual assuming 8 hour work days, 5 days a week, and 50 weeks per year.

[c] Water requirements are for the aqueous processing of feedstock material. Other processes would require less.

Source: SRNS 2020.

Nonradiological Releases

Nonradiological emissions for feedstock preparation would be associated with the transport of material to the K-Reactor Building and worker vehicles. Emission data is presented in **Table B–45**.

Table B–45. Savannah River Site Feedstock Preparation Facility Annual Operational Nonradiological Emissions

Facility	Emissions (tons)							Combined HAPs [a]	CO_2e (mt)
	VOC	CO	NOx	SO2	PM_{10}	$PM_{2.5}$	CO_2		
Onsite Emissions from On-road Sources	0.02	0.23	0.03	0.0003	0.001	0.001	46	0.002	42
Onsite Emissions from Nonroad Sources	0.002	0.01	0.03	0.0001	0.004	0.001	16	0.0004	15
Offsite Emissions from On-road Sources	0.07	7.58	0.39	0.007	0.19	0.04	1,000	0.02	909
Total 2025 Emissions	**0.08**	**7.82**	**0.45**	**0.01**	**0.20**	**0.04**	**1,062**	**0.02**	**965**

CO = carbon monoxide; CO_2 = carbon dioxide; CO_2e = carbon dioxide equivalent; HAPs = hazardous air pollutants; MT = metric tons; NO_x = nitrogen oxides; $PM_{2.5}$ = particulate matter less than 2.5 microns in diameter; PM_{10} = particulate matter less than 10 microns in diameter; SO_2 = sulfur dioxide; VOC = volatile organic compound.

[a] Combined HAPs = 15/3 percent of combustive VOC/PM emissions for on-road and nonroad sources and 1/3 percent for slash burning (California Air Resources Board 2018).

Source: Adapted from SRNS 2020.

Radiological Releases

Radiological releases for feedstock preparation at SRS would be the same as described for that activity at the INL Site. See Table B–34 in Section B.5.3.1.3.

HEPA-filtered releases of radioactivity to the environment would be through a stack installed for the driver fuel fabrication facility. The combined flow rate would be about 18,000 cubic feet per minute at an elevation of about 124 feet (SRNS 2020).

Waste Generation

Annual waste generation rates, based on the steady state production of about 45 driver fuel assemblies per year are provided in **Table B–46**. Estimated waste quantities for production (feedstock preparation and fuel fabrication) have been developed without considering any potential reduction in wastes that would result from the performance of both processes. In particular primary transuranic waste would not be doubled if both feedstock preparation and fuel fabrication were to be required. Estimated waste also may vary with the quality of the plutonium feedstock. The quantities listed here are expected to be representative of the waste generated during feedstock preparation.

Table B–46. Savannah River Site Annual Feedstock Preparation Facility Operational Wastes

Waste	*Volume (cubic meters)*
Low-level radioactive	170
Mixed low-level radioactive [a]	2
Secondary transuranic	32
Secondary mixed transuranic [a]	10
Primary transuranic	170
Hazardous – solid	1
Hazardous – liquid	1
Nonhazardous – solid	17
Nonhazardous – liquid	200
Universal	0.42

[a] For low-level and secondary transuranic radioactive wastes, the mixed waste volumes are included in the total waste volume.

Source: SRNS 2020.

B.5.4.2 Savannah River Site Fuel Fabrication

Under the SRS fuel fabrication option, driver fuel fabrication would be performed in the K-Reactor Building (105-K) in the K Area Complex. All equipment necessary to support fuel alloying and homogenization, fuel slug casting, fuel pin assembly, and driver fuel assembly fabrication would be located on two below-ground levels within the building.

Under the SRS Fuel Processing and Conversion Option, this capability would be located adjacent to the location for the fuel fabrication capability, in the K-Reactor Building (105-K) in the K Area Complex. All of the equipment for fuel processing and conversion would be newly constructed.

B.5.4.2.1 Savannah River Site Fuel Fabrication Facilities Overview

At SRS, the fuel fabrication facility would be located on the minus-20- and minus-40-foot levels (20 and 40 feet below grade) of the K-Reactor Building, Building 105-K. The facility is located within a Protected Area and includes a Material Access Area with the physical security infrastructure that satisfies requirements for handling and storage of Category I special nuclear material. This area is currently used to store drums of heavy water and pumps (SRNS 2020).

Approximately 17,000 square feet and 22,600 square feet of space would be made available at the minus-40- and minus-20-levels, respectively. Material and equipment to be removed are expected to be radiologically clean. A portion of the space at the minus-20-foot level has a high bay area that would allow for the vertical assembly of driver fuel assemblies. The identified area would be suitable for the fuel fabrication glovebox processes being designed at the INL Site. The facility could support feed material purification, ingot manufacturing, and/or the fabrication of fuel from ingots. New equipment would be provided for fuel slug casting, slug trimming and inspection, fuel rod loading and inspection, fuel bundle assembly and packaging, and waste handling. Other infrastructure to be supplied would include material storage areas (including an area to store fully assembled driver fuel assemblies), special nuclear material measurement equipment, analytical support, and other infrastructure services such as glovebox and room ventilation and electrical distribution (SRNS 2020).

The facility design would be based on the conceptual design developed for the fuel fabrication facility at the INL Site. While a specific layout has not been established, the following is a notional layout to convey the type and size of equipment and the representative space needed for operations. Structural modifications to the facility would be required to accommodate fuel fabrication. At SRS, fuel ingots would be received at ground level and transferred via an existing, but to be upgraded, elevator to a small lag vault located in one of the motor rooms at the minus-40-foot level. Two process lines for alloy mixing, slug casting, and pin assembly would be located at the -40-foot level within the existing Cross-over Area and the Process and Pump Rooms (see **Figure B–29**). Additionally, equipment for fuel pin non-destructive analysis, waste processing,[25] and analytical support would be located at this level. Assembled fuel pins would be transferred to a high bay area at the minus-20-foot level for preparation and assembly into complete driver fuel assemblies (see **Figure B–30**). (Alternately final assembly could be done in the K-108 Building or at the -40-foot level (provided some heat exchangers were removed from this area). Since SRS is not a proposed site for the VTR, completed assemblies would be loaded into a shielded transfer cask at the minus-20-foot Assembly Area Basement. The shielded transfer cask would be raised up out of the Assembly Area Basement and then loaded into a shipping container for shipment (SRNS 2020).

Although the VTR modifications have not been designed, based on similar K Area upgrade projects, the space needed for support facilities for the needed HVAC, fire suppression, etc. are expected to be substantial. At least one and possibly two, of the adjacent 108-K buildings could be needed for these support operations. The addition of a new 8,000-square foot structure to house an upgraded HVAC system would be required. This structure could be contained within one of the 108-K buildings, placed on top of one of the buildings or placed on a previously disturbed area (less than 3 acres) within the K-Reactor Building security area, depending on whether one or both of feedstock preparation and fuel fabrication were to be located at SRS. (This is the same HVAC capability described under SRS feedstock preparation.) Additional modifications could include construction of a new facility stack (the preconceptual design includes a 124-foot stack) for the VTR fuel production activities and construction of a new entry control structure.

Should SRS be selected as the site for fuel fabrication, a demonstration facility would still be built at INL. The demonstration facility would be located in the existing INL FMF at the same location as the proposed production facility. It would consist of a single line of furnace, demolding, and pin-assembly gloveboxes. Scrap processing, waste handling, and fuel slug quality assurance gloveboxes would also be installed.

[25] Scrap unsuitable for reuse would be transferred to the oxidation/blenddown line where the alloy would be oxidized and blended down to meet the Waste Isolation Pilot Plant facility disposal and safeguards and security criteria.

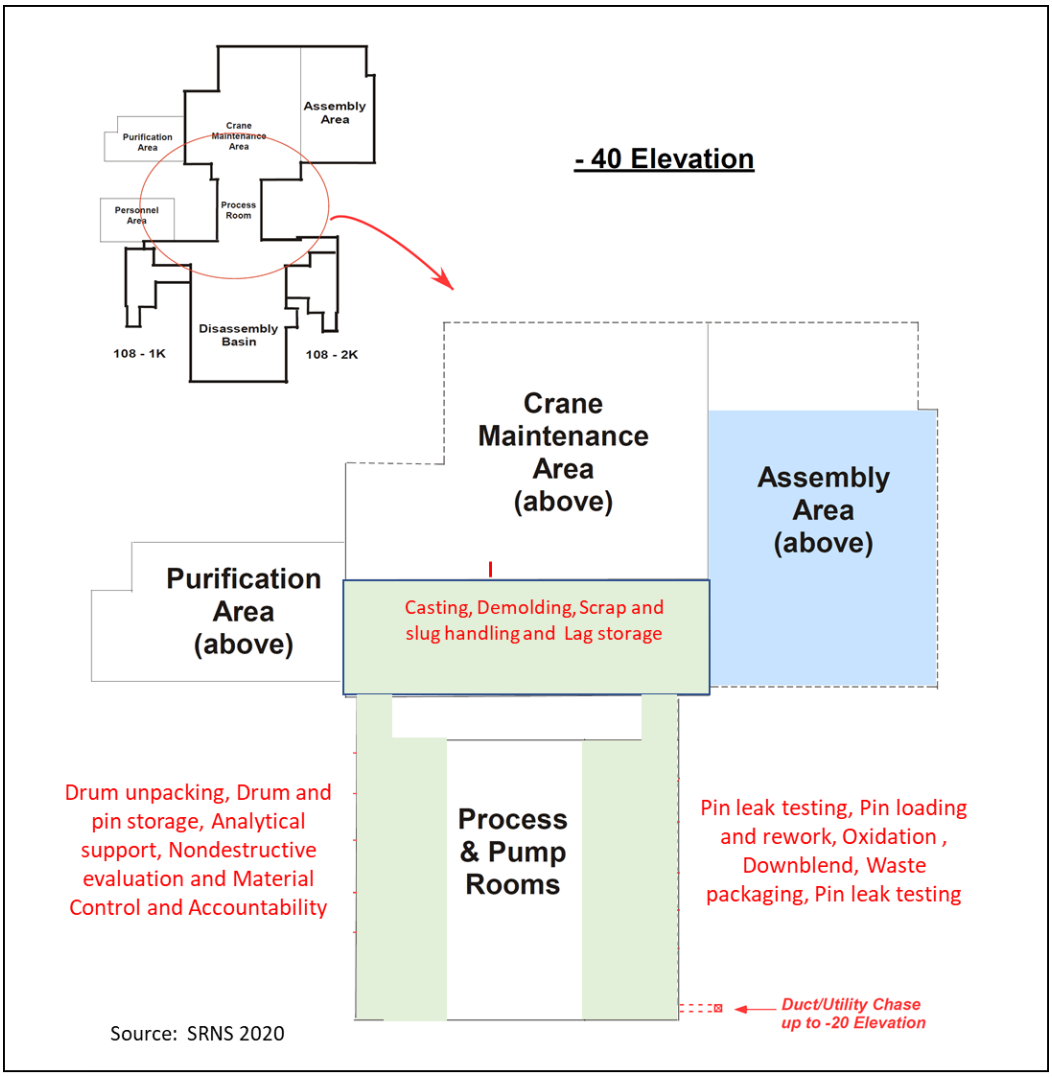


Figure B–29. Savannah River Site Proposed Fuel Fabrication Facility Minus-40-Foot Level of K-Reactor Building

B.5.4.2.2 Environmental Resources – Construction

Metallic feed stock would be delivered to the K-Reactor Building (K-105), and no new facilities would be constructed at SRS. The only construction activities would be the build-out of the equipment locations within K-Reactor Building and the removal of existing equipment. Construction is assumed to require 3 years. A few (three) small, previously disturbed areas, totaling less than an acre) within the K-105 security fencing have been identified as potential construction laydown areas.

Resource Requirements

Table B–47 provides a summary of the key resources committed to the modification of the K-Reactor Building to enable its use as a fuel fabrication facility. In addition to the materials identified in this table, materials used in the construction of the gloveboxes include stainless steel for structural supports, glass for glovebox windows, piping for inlet, exhaust and other gas lines, electrical cable, and conduit for power and instrument lines. Primary gases used in the gloveboxes include argon as an atmosphere and hydrogen as a mechanism to remove oxygen from the glovebox atmosphere.

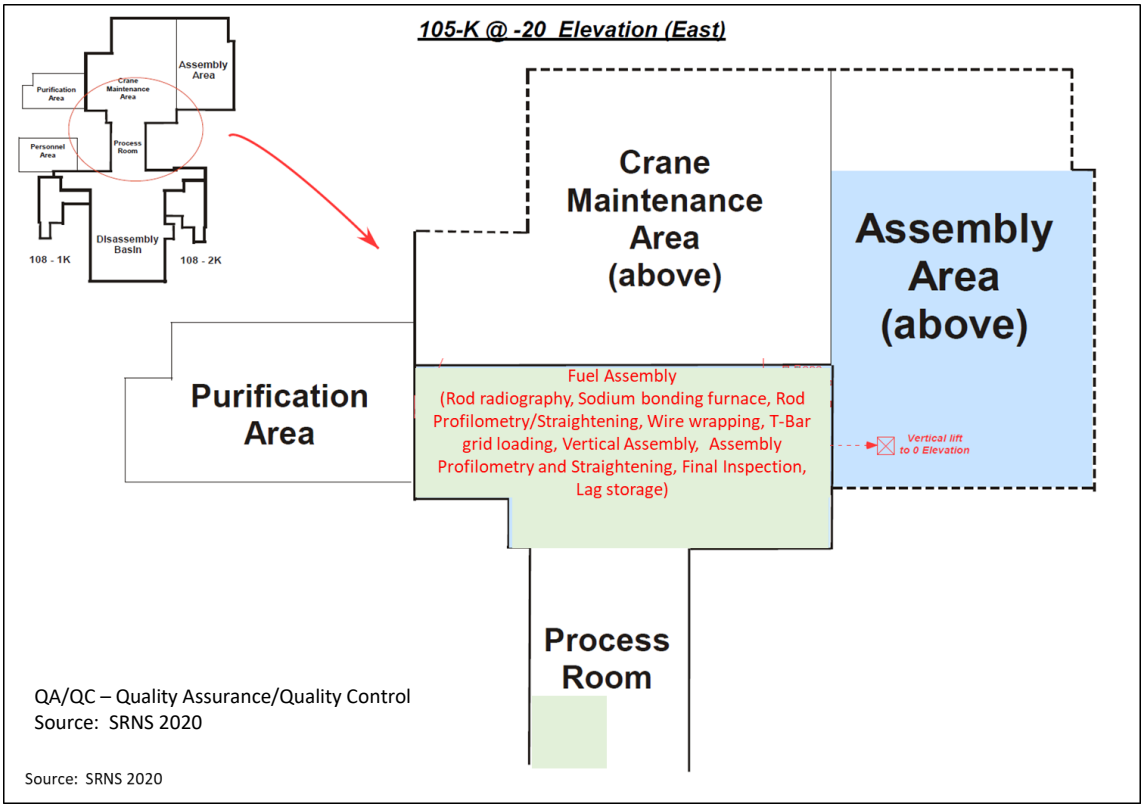


Figure B–30. Savannah River Site Proposed Fuel Fabrication Facility Minus-20-Foot Level of K-Reactor Building

Table B–47. Savannah River Site Fuel Fabrication Facility Construction Resource Requirements

Resource	*Units*	*Value*	
		Annual Average	*Total* [a]
For Modifications to Existing Facilities			
Staff	FTE	120	360
Electricity	kWh	Minimal	Minimal
Diesel Fuel	gallons	1,500	4,500
Gasoline	gallons	2,500	7,500
Water [b]			
Potable	gallons (thousands)	1,000	3,000
Construction	gallons (thousands)	2,000	6,000
Total	gallons (thousands)	3,000	9,000
Construction Materials			
Cement	tons	--	800
Steel (tons)	tons	--	600
Conduit	linear feet	--	74,000
Cable Tray	linear feet	--	2,500
Power/Control Cable	linear feet	--	83,000
Piping	linear feet	--	14,000
Facilities	square feet	--	40,000
Ductwork	pounds	--	51,000
Formwork	square feet	--	36,000
Sand, Cone, Aggregate	cubic yards	--	880
Gravel, Crushed Stone, etc.	cubic yards	--	660
Soil – Fill Material	cubic yards	--	3,600

Resource	Units	Value	
		Annual Average	Total [a]
Gases			
Acetylene	cubic meters	--	53
Oxygen	cubic meters	--	240
CO_2/Argon	cubic meters	--	80
Nitrogen	cubic meters	--	160
Argon	cubic meters	--	1,300
Helium	cubic meters	--	33
Other			
Epoxy Floor Covering	square feet	--	48,000
Macropoxy (concrete wall covering)	square feet	--	117,000
Enamel Paint	square feet	--	50,000
Intumescent Coating (steel deck coating)	square feet	--	8,300

CO_2 = carbon dioxide; FTE = full-time equivalent (person); kWh = kilowatt-hour.

[a] A 3-year construction period.

[b] Water use provided as gallons per minute, converted to annual assuming 10-hour work days, 5 days a week, and 50 weeks per year and is based on the peak construction workforce.

Source: SRNS 2020.

Nonradiological Releases

Nonradiological releases are associated with the operation of the forklifts, construction vehicles, concrete mixers, cranes and other smaller equipment (i.e., the burning of diesel fuel and worker personal vehicle use). The annual emissions associated with these items would be about the same as those associated with feedstock preparation (see Table B–42).

Waste Generation

Table B–48 provides waste generation information for construction of the fuel fabrication facility. Wastes associated with construction activities would be comprised of three main types: obsolete or replaced equipment, radiologically contaminated construction wastes, and cleaning supplies and clean wastes.

Table B–48. Savannah River Site Fuel Fabrication Facility Construction Wastes

Waste Type	Units	Value
Toxic Substance Control Act Waste	cubic meters	28
Universal Waste	cubic meters	7.5
Nonhazardous Waste		
From Construction Activities	gallons/cubic meters	90,000/340
Equipment Removed	metric tons/cubic meters	100/5,000
Low-level Radioactive Waste	cubic meters	770

Source: SRNS 2020.

The majority of the dismantlement and removal (D&R) items to be removed from the minus-40-foot motor rooms and crossover and the minus-20-foot pipe corridors and crossover are expected to be nonradioactive. There are a few contamination areas that have the potential to generate low-level radioactive waste (LLW). Radiological control operations personnel will be involved in determining which items can be free released, which items fall under the metals moratorium, and which items may have to be treated as LLW due to unknown history. In addition, all items will require evaluations for asbestos, polychlorinated biphenyls and Resource Conservation and Recovery Act constituents prior to determining a final disposition path (SRNS 2020).

It is anticipated that asbestos will be encountered during D&R activities. An inspection will be conducted by a licensed inspector prior to initiation of D&R activities and as needed during D&R when suspect

materials are encountered to properly identify asbestos-containing materials and presumed asbestos-containing materials (SRNS 2020).

Although detailed estimates of the decontamination and decommissioning waste are not available, the mass of the removed material could be as high as 100 metric tons and 5,000 cubic meters in packaged form[26] (SRNS 2020). This material would be disposed at either onsite LLW sites or onsite construction and demolition landfill disposal sites.

B.5.4.2.3 Environmental Resources – Operations

The fuel fabrication facility would produce up to 19,530 usable fuel slugs per year when each fuel pin contains two fuel slugs, sufficient to supply up to 45 fresh driver fuel assemblies per year. A portion of the fuel slugs produced would not be expected to meet VTR fuel requirements. Most of the unusable fuel slugs could be processed in the feedstock preparation facility and would be recast into fuel slugs. However, some of the material would be expected to be captured in one of the fuel fabrication waste streams.

Should SRS be selected as the site for fuel fabrication, a demonstration fuel fabrication line would be built at INL. Environmental resources associated with the operation of this demonstration line for the full duration of its operation would be bound by the resources associated with one year of operation of the INL fuel fabrication facility. These operational environmental resources are discussed in Section B.5.3.2.3.

Resource Requirements

Key annual resource commitments for the operation of the fuel fabrication facility are provided in **Table B–49**. Resource requirements listed do not include the fuel fabrication material (uranium, plutonium, zirconium, sodium, and HT-9 stainless steel)

Table B–49. Savannah River Site Annual Fuel Fabrication Facility Resource Requirements

Resource	*Units*	*Value*
Staff	FTE	300
Electricity	MWh	8,300-13,300 [a]
Diesel		
Centerra [b]	gallon	3,000
Operations [b]	gallon	4,000
Total	gallon	7,000
Water Supply [c]	gallons (thousands)	1,400
Wastewater Treatment	gallons (thousands)	1,400
Argon	cubic meters	600,000
helium	cubic meters	30,000
Nitrogen	cubic meters	30,000
Oxygen	cubic meters	30,000
Propane	bottles/gallons	100/470
Quartz	kilograms	3,000
Ytrria	kilograms	9
Zirconia Mold Wash	kilograms	90
Graphite	kilograms	500

FTE = full-time equivalent (person); MWh = megawatt-hour.

[a] High and low of estimated values.

[b] Diesel fuel for one additional security vehicle (Centerra) and an additional diesel generator (Operations).

[c] Water use provided as gallons per minute, converted to annual assuming 8-hour work days, 5 days a week, and 50 weeks per year.

Source: INL 2020c; SRNS 2020.

[26] If the heat exchangers are removed from the minus-40-foot level, an additional 18 truckloads of debris would be generated.

Nonradiological Releases

Nonradiological emissions for fuel fabrication would be associated with the transport of material to the K-Reactor Building and worker vehicles. Emission data would be similar to that for INL feedstock preparation, see Table B–45.

Radiological Releases

HEPA-filtered radiological releases would be the same as for fuel fabrication at INL. See Section B.5.3.2.3, Table B–39.

Releases of radioactivity to the environment would be through a stack installed for the VTR fuel fabrication facility or an existing stack. The combined flow rate would be about 18,000 cubic feet per minute at an elevation of about 124 feet (SRNS 2020).

Waste generation

Annual waste generation rates, based on the production of about 45 driver fuel assemblies per year are provided in **Table B–50**. The rates shown in the table are for the fabrication of fuel directly from feedstocks; feedstocks for which no feedstock preparation would be required. These feedstocks would contain impurities at levels below the acceptable limits for the VTR fuel. Should feedstock preparation be required, the transuranic wastes generated from fuel fabrication would be much less than the values shown in Table B–50. Other wastes would be generated in quantities similar to those shown.

Table B–50. Savannah River Site Annual Fuel Fabrication Facility Operational Wastes

Waste	*Volume (cubic meters)*
Low-level radioactive	170
Mixed low-level radioactive [a]	2
Secondary transuranic	32
Secondary mixed transuranic [a]	10
Primary transuranic	170
Hazardous – solid	1
Hazardous – liquid	1
Nonhazardous – solid	17
Nonhazardous – liquid	200
Universal	0.42

[a] For low-level and secondary transuranic radioactive wastes, the mixed waste volumes are included in the total waste volume.

Source: SRNS 2020.

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